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NUREG-2266

Environmental Evaluation of Accident Tolerant Fuels with Increased Enrichment and Higher Burnup Levels

Final Report

Office of Nuclear Material Safety and Safeguards

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Environmental Evaluation of Accident Tolerant Fuels with Increased Enrichment and Higher Burnup Levels

Final Report

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ABSTRACT

When reviewing a license amendment request (LAR) to adopt accident tolerant fuel (ATF) with increased enrichment and higher burnup levels beyond the currently licensed limits, the U.S. Nuclear Regulatory Commission (NRC) staff will need to evaluate the potential environmental impacts of the request. Conducting complete environmental evaluations for each individual site could result in unnecessarily complex and lengthy assessments of onsite and offsite environmental impacts. While some environmental impacts from the deployment and use of ATF will be dependent on site- and design-specific safety considerations, such as radiological effluent releases and postulated accidents, the conditions common to all light-water reactors (LWRs) for other environmental impacts could be beyond previous LWR environmental evaluations. Specifically, the anticipated enrichment levels above 5 weight percent (wt%) of uranium-235 (U-235) and burnup levels above 62 gigawatt days per metric ton of uranium (GWd/MTU) are outside the conditions supporting Table S-3 of Title 10 of the *Code of Federal Regulations* [10 CFR] Section 51.51(b) (10 CFR 51.51(b)) for uranium fuel cycle environmental impacts and the conditions for the use of Table S-4 of 10 CFR 51.52(c) regarding fuel and waste transportation environmental impacts, and could affect the level of environmental impacts during decommissioning.

To support efficient and effective licensing reviews of ATFs and to reduce the need for a complex site-specific environmental review for each ATF LAR, this NUREG evaluated the reasonably foreseeable impacts of near-term ATF technologies (both first- and second-generation) with increased enrichment and higher burnup levels on the uranium fuel cycle, transportation of fuel and waste, and decommissioning for LWRs (i.e., a bounding analysis). To this end, the NRC staff assessed and applied available near-term ATF technology performance analyses, data, and studies; information from prior NRC environmental analyses; and the assessment of other publicly available data sources and studies to complete an evaluation of ATF with increased enrichment and higher burnup levels. Based on the evaluations in this study, Table S-3 of 10 CFR 51.51(b) with the *Continued Storage Generic Environmental Impact Statement,* and *the Decommissioning Generic Environmental Impact Statement* would bound

the deployment and use of near-term ATF for up to 10 wt% U-235 and up to 80 GWd/MTU average assembly burnup. Table S-4 of 10 CFR 51.52(c) would bound the deployment and use of near-term ATF for up to 8 wt% U-235 and 80 GWd/MTU average assembly burnup. This study also indicates there would be no significant adverse environmental impacts for the uranium fuel cycle, transportation of fuel and wastes, and decommissioning associated with deploying near-term ATF.

Paperwork Reduction Act Statement

This NUREG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Part 51 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) under control number 3150-0021. Send comments regarding these information collections to the Freedom of Information Act, Library, and Information Collections Branch (T6A10M), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, or by email to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0021). Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street NW, Washington, DC 20503.

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EXECUTIVE SUMMARY

To support efficient and effective licensing reviews of new accident tolerant fuels (ATFs) and to reduce the need for a complex site-specific environmental review for each ATF license amendment request, this study evaluated the likely impacts of near-term ATF technologies with increased enrichment and higher burnup levels on the uranium fuel cycle, transportation of fuel and waste, and decommissioning of light-water reactors (LWRs) (i.e., a bounding analysis). Near-term ATF technologies are coated cladding and doped pellets as first-generation technologies, and iron-chromium-aluminum (FeCrAl) cladding as a second-generation technology. Other long-term ATF technologies are not a part of this study. The U.S. Nuclear Regulatory Commission (NRC) staff evaluated the impact of increased enrichment and higher burnup levels by assessing and applying NRC-sponsored ATF technology reports, prior environmental reviews, transportation studies, and new or updated data sources to determine the bounding (generic) environmental impacts of deploying ATF technologies with increased enrichment and higher burnup levels in LWRs.

The NRC initially considered the environmental impacts of the uranium fuel cycle in WASH-1248. There have been significant changes in the front-end processes and NRC-licensed facilities since the publication of WASH-1248. The most notable examples of these changes are the extraction of uranium from the ground using in situ recovery instead of traditional mining, performance of all enrichment with gaseous centrifuges instead of gaseous diffusion, and electricity generation moving significantly away from the use of coal. The result of these various changes is to significantly reduce the environmental effects of the front-end of the uranium fuel cycle. Thus, the environmental effects of the front-end of the uranium fuel cycle from the deployment and use of ATF with increased enrichment for up to 10 weight percent (wt%) uranium-235 (U-235) are bounded by the environmental effects provided in Table S-3 of 10 CFR 51.51.

Regarding the back-end of the uranium fuel cycle, the current practice of long-term storage and management of spent nuclear fuel (SNF) would still apply to the deployment and use of ATF with increased enrichment and up to 80 gigawatt days per metric ton of uranium (GWd/MTU) assembly average burnup levels. Consistent with NRC regulations and thermal loading requirements for licensed spent fuel storage cask systems, specific cooling times in a spent fuel pool would be necessary prior to transferring the spent fuel to an Independent Spent Fuel Storage Installation (ISFSI).

A benefit of the deployment and use of ATF with increased enrichment and higher burnup levels would be the longer times between refueling operations, which would lessen the average annual rate at which licensees place spent ATF assemblies into the spent fuel pools and ultimately transfer spent ATF assemblies to an ISFSI relative to the rate for traditional spent fuel. This could, in turn, lessen the overall amount of SNF stored at a site and lengthen the time before licensees need to expand an ISFSI relative to facilities using fuel that have lower enrichments and lower burnup levels. This lessens the environmental impacts compared to what would occur with current fuel, which would be consistent with prior NRC environmental evaluations. Spent ATF storage would be consistent with earlier published analyses, would not require any significant departure from certified spent fuel shipping and storage containers, and would continue under an approved aging management program.

When conducting the generic analysis in the Continued Storage Generic Environment Impact Statement (GEIS), the NRC staff applied conditions and parameters that are sufficiently conservative to bound the impacts such that any variances that might occur from site to site are unlikely to result in environmental impact determinations that are greater than those presented in the Continued Storage GEIS. Therefore, with respect to ATF storage, including spent ATF with increased enrichment and higher burnup levels, the period beyond the licensed life for operation of a reactor, spent ATF would conform with the analysis of the Continued Storage GEIS, and accordingly the Continued Storage GEIS would bound the impacts from deployment and use of ATF.

The analysis of the transportation of ATF and ATF waste with increased enrichment and higher burnup levels is based on shipment of low-level radioactive waste and unirradiated, and spent ATF, including those with increased enrichments and higher burnup levels, by legal weight trucks in certified transport packages. The transportation impacts are divided into two parts. The first part considers normal conditions, or incident-free, transportation, and the second part considers transportation accidents.

Shipments that take place without the occurrence of accidents are routine, incident-free shipments and the radiation doses to various receptors (exposed persons) are called incident-free doses. The vast majority of radioactive shipments are expected to reach their destination without experiencing a transportation accident or incident or releasing any cargo (to date, there have been no shipments of spent fuel resulting in a release of radioactive material to the environment). As previously noted, deployment and use of ATF with increased enrichment and higher burnup levels could result in the lengthening of the time between refueling operations, leading to an overall reduction of the number of spent fuel assemblies needing to be shipped offsite on an annual basis. Such reduction would lessen the environmental impacts compared to what would occur with current fuel and refueling operations due to transportation of spent fuel. The incident-free impacts from these normal, routine shipments arise from the low levels of radiation that are emitted externally from the shipping container.

Incident-free legal weight truck transportation of spent ATF, including spent ATF with increased enrichment and higher burnup levels, has been evaluated by considering shipments from six representative LWR sites to a postulated permanent geological repository for SNF in the western United States.[1] As a surrogate for such a postulated permanent geologic repository, the NRC has used the proposed Yucca Mountain, Nevada site for the transportation analysis. The six LWR sites from which the shipments originate include the following:

- Brunswick Steam Electric Plant (Brunswick);
- Columbia Generating Station (Columbia);
- Dresden Nuclear Power Station (Dresden);
- Enrico Fermi Nuclear Generating Station Unit 2 (Fermi);
- Millstone Power Station (Millstone); and
- Turkey Point Nuclear Plant (Turkey Point).

For each LWR site, the NRC staff considered and evaluated both boiling water reactor (BWR) and pressurized water reactor (PWR) spent ATF shipments, including ATF with increased enrichment and higher burnup levels, for the purpose of impact comparison owing to the different release fractions for BWR and PWR fuel designs.

[1] Assuming a western repository location ensures distances for transportation routes and the associated impacts are not underestimated given the locations for most LWR sites are in the eastern portion of the United States.

Environmental impacts from these shipments would occur to persons residing along the transportation corridors between the reactor sites and the repository, to persons in vehicles passing the spent fuel shipments in the same and opposite directions, to persons at vehicle stops (such as rest areas, refueling stations, inspection stations, etc.), and to transportation crew members. For the purposes of this analysis, the transportation crew for truck spent fuel shipments consisted of two drivers. The regulatory maximum crew dose rate of 2 millirem per hour (mrem/hr), and regulatory maximum transport package surface dose rate of 10 mrem/hr at 2 meters is conservatively used in the analysis. The characteristics of specific shipping routes (e.g., population densities, shipping distances) influence the normal radiological exposures.

The accident risks are the product of the likelihood of an accident involving a spent fuel shipment and the consequences of a release of radioactive material resulting from the accident. The likelihood of an accident is directly proportional to the number of fuel shipments. Accident risks also include a consequence term. Consequences are represented by the population dose from a release of radioactive material given that an accident occurs that leads to a breach in the shipping package's containment systems. Consequences are a function of the total amount of radioactive material in the shipment, the fraction that escapes from the shipping package, the fraction of the release from the shipping package that is aerosolized, the fraction of the release that is respirable, the dispersal of radioactive material to humans, and the characteristics of the exposed population. The NRC staff used the shipping distances and population distribution information for the regions pertaining to the sites used for the evaluation of the impacts of incident-free transportation for accident impact evaluations. The NRC staff used the most recent available data on accident rates, release fractions, aerosolized fractions, and respirable fractions in this evaluation.

The transportation impact evaluation includes the use of the NRC-maintained NRC-Radioactive Material Transport (NRC-RADTRAN) transportation risk code package, pertinent fuel radionuclide inventory (source term) data, and external and accidental release characteristics, routing distance information, and population density by State along the route. The staff obtained routing information by running the Web-Based Transportation Routing Analysis Geographic Information System (WebTRAGIS) code. While the population density considered in WebTRAGIS is for the year 2012, based in part on the 2010 U.S. Census data, the staff extrapolated the population density to 2022 based on each State's growth rate using 2010 and 2020 U.S. Census data. The staff compiled information with respect to vehicle daily traffic count, vehicle speed, vehicle accident, fatality, and injury rates from U.S. Department of Transportation data base and used that information in the NRC-RADTRAN analysis to determine single shipment impacts. To determine annual transportation impacts, the staff applied the normalized (annual) truck shipments of 52 shipments and 30 shipments estimated spent ATF from a BWR and PWR, respectively.

The NRC staff found the maximum normal conditions (i.e., incident-free) cumulative worker dose per year (yr) was bounded by the 4 person-rem per reactor-year value of Table S-4 of 10 CFR 51.52 (TN250). This worker dose would be managed with multiple drivers available as the transportation crew so that the individual worker dose would be below the U.S. Department of Energy administrative limit of 2 rem/yr and the NRC's occupational exposure annual limit of 5 rem/yr. PWR shipment cumulative public doses were at or slightly higher than the 3 person-rem/yr specified in the Table S-4. The NRC staff found the cumulative public dose per year for the BWR shipments to be higher than 3 person-rem/yr, but both the BWR and PWR results are not significant when the related average individual dose is considered. Namely, the average individual doses along all routes and fuel types are well below 1 mrem/yr, a small fraction of the average annual natural background radiation exposure of approximately 310 mrem, and within the Table S-4 range of doses to exposed individuals. These results are conservative because they are based on the transport package that has the least capacity. Applying a transport package with a greater capacity would reduce the number of shipments resulting in a lower cumulative dose that would be less than the 3 person-rem of Table S-4, as shown by the rail sensitivity case in this study (e.g., the GA-4 truck spent fuel transport can hold four PWR fuel assemblies, which would reduce the PWR cumulative doses by a factor of 4).

The NRC staff found that the total accidental population risk per year due to transport of spent ATF, including spent ATF with increased enrichment and higher burnup levels, continued to demonstrate the low risks from both radiological and nonradiological accidents and is consistent with past transportation studies. The greater risk to a member of the public would be physical harm from an actual vehicle collision involving a spent ATF shipment, if such an event ever happens. While the nonradiological risk is the greater risk, the results of this study demonstrate that such risks would still not be significant and are less than the common (nonradiological) cause environmental risks of Table S-4. The results for spent ATF with increased enrichment and higher burnup levels are consistent with the environmental impacts associated with the transportation of fuel and radioactive wastes to and from current-generation reactors presented in Table S-4 of 10 CFR 51.52 (TN250).

Based on the results of the impact analysis, shipments of near-term ATF technologies (first- and second-generation) with enrichments of up to 8 wt% uranium-235 (U-235) and higher assembly averaged burnup levels of up to 80 GWd/MTU would not significantly change the potential impacts of either incident-free or accident transportation risk. Hence, the impact of transporting spent ATF is bounded by Table S-4. Therefore, the results of this analysis could serve as a reference in helping to address the environmental impacts of ATF licensing without a detailed site-specific transportation analysis, as long as the ATF is within the enrichment and burnup levels of the associated fuel assembly radionuclide inventory and parameters applied in the analyses of this NUREG.

In the case of decommissioning, the expected impacts from deployment and use of ATF with increased enrichment and higher burnup levels would be the same as or slightly less than those from decommissioning nuclear power plants operating with the existing fuel. Additionally, the expected Decommissioning GEIS and guidance updates could build upon the analysis from this study to specifically address the decommissioning of a LWR deploying and using ATF.

The NRC staff concludes that the findings in this NUREG addressing near-term ATF technologies indicate the environmental effects associated with deploying and using ATF would be bounded by the NRC staff’s prior analysis related to Tables S-3 and S-4, the Continued Storage GEIS, and the Decommissioning GEIS. For enrichments up to 10 wt% U-235 and assembly averaged burnup of up to 80 GWd/MTU, the analysis in this NUREG bounds the environmental effects of ATF with respect to the uranium fuel cycle and decommissioning. For the transportation of fuel and waste, the analysis in this NUREG bounds the environmental impact of ATF for enrichments up to 8 wt% U-235 and assembly averaged burnup up to 80 GWd/MTU. Additionally, this NUREG could provide guidance for completing the needed revised analysis in a future licensing action if the enrichment and burnup levels are greater than those stated above or for the deployment and use of long-term ATF technologies.

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ACRONYMS AND ABBREVIATIONS

°C	degree(s) Celsius
°F	degree(s) Fahrenheit
µm	micron(s)
ADOPT	Advanced Doped Pellet Technology
ADU	ammonium diuranate
AEC	U.S. Atomic Energy Commission
Am	americium
ATF	accident tolerant fuel
atm	atmosphere
Brunswick	Brunswick Steam Electric Plant
BWR	boiling water reactor
CFR	*Code of Federal Regulations*
CO_2e	carbon dioxide equivalent
Co-60	cobalt-60
Columbia	Columbia Generating Station
CSAS	Criticality Safety Analysis Sequence
Cs	cesium
Ci	curie(s)
Cm	curium
DECON	decontamination
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
Dresden	Dresden Nuclear Power Station
DTS	Dry Transfer System
E	Within this document, scientific notation is denoted by E followed by the exponent. For example, 5.02E-02 denotes 5.02×10^{-2} and 5.02E+02 denotes 5.02×10^{2}
EA	environmental assessment
EISs	environmental impact statements
ENTOMB	entombment
ENDF	Evaluated Nuclear Data File
FeCrAl	iron-chromium-aluminum
Fermi	Enrico Fermi Nuclear Generating Station
FMCSA	Federal Motor Carrier Safety Administration

Framatome FFF	Framatome, Inc. Fuel Fabrication Facility
ft	foot/feet
ft^3	cubic foot/feet
GEIS	*Generic Environmental Impact Statement*
GUI	graphical user interface
GWd/MTU	gigawatt day(s) per metric ton of uranium
HAZMATs	hazardous materials
HBU	higher burnup
He	helium
HLW	high-level waste
IAEA	International Atomic Energy Agency
in	inch(es)
ISFSI	Independent Spent Fuel Storage Installation
K	Kelvin
kg	kilogram(s)
km	kilometer(s)
kW	kilowatt(s)
kWh	kilowatt-hour(s)
Kr	krypton
Kr-85	krypton-85
LAR	license amendment request
lb	pound(s)
LEU	low enriched uranium
LEU+	low enriched uranium plus
LLRW	low-level radioactive waste
LWR	light-water reactor
m	meter(s)
m^3	cubic meter(s)
mi	mile(s)
mol	mole(s)
mph	mile(s) per hour
mrem/hr	millrem per hour
MT	metric ton(s)
MTU	metric ton(s) of uranium
MTU/day	metric ton(s) of uranium per day
MTU/year	metric ton(s) of uranium per year

MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NAC-LWT	NAC International-Legal Weight Truck
NAC-STC	NAC-Storage Transport Cask
NEIMA	Nuclear Energy Innovation Modernization Act of 2019
NIMA	National Imagery and Mapping Agency
NPDES	National Pollutant Discharge Elimination System
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PNNL	Pacific Northwest National Laboratory
PSDAR	post-shutdown decommissioning activity report
Pu	plutonium
PWR	pressurized water reactor
RADTRAN	Radioactive Material Transport
RAMP	Radiation Protection Computer Code Analysis and Maintenance Program
Rb	rubidium
rem/yr	Roentgen equivalent man per year
Ru	ruthenium
SAFDL	specified acceptable fuel design limit
SAFSTOR	SAFe STORage
Sandia	Sandia National Laboratories
SiC	silicon carbide
SNF	spent nuclear fuel
Sr	strontium
SWU	separative work unit
TIGER	Topologically Integrated Geographic Encoding and Referencing (system)
Turkey Point	Turkey Point Nuclear Generating Units 3 and 4
U-235	uranium-235
U-238	uranium-238
UF_6	uranium hexafluoride
UN	uranium nitride
UO_2	uranium dioxide
U_3O_8	triuranium octaoxide
WebTRAGIS	Web-Based Transportation Routing Analysis Geographic Information System

wt%	weight percent
Xe	xenon
Y	yttrium

1 INTRODUCTION

1.1 Purpose for this Study

The U.S. Nuclear Regulatory Commission (NRC) staff is preparing to review applications related to the deployment of new accident tolerant fuel (ATF) technologies (i.e., fuels with longer coping times during loss-of-cooling conditions) in U.S. commercial light-water reactors (LWRs) (NRC 2021–TN8017). The NRC staff is anticipating license amendment requests (LARs) for the deployment and use of ATF technologies in LWRs, each requiring a separate environmental review to meet the agency's National Environmental Policy Act (NEPA) obligation (see Title 10 of the *Code of Federal Regulations* [10 CFR] Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions; Subpart A, NEPA–Regulations Implementing Section 102(2) TN250).

During an environmental review, the NRC staff must evaluate a range of environmental considerations for deployment and use of ATF technologies in LWRs. Several of the environmental considerations are common across all LWRs and can be assessed prior to deployment of ATF technologies. The common environmental considerations assessed by this study involve increased enrichment and higher burnup spent fuel management in the uranium fuel cycle, transportation of fuel and waste to and from a nuclear power plant (NPP), and LWR decommissioning.

1.2 Background

On January 14, 2019, the President signed the Nuclear Energy Innovation and Modernization Act (NEIMA 2019-TN6469). The NEIMA, Section 107, "Commission report on accident tolerant fuel," defines ATF as a new technology that does the following:

- "makes an existing commercial nuclear reactor[[1]] more resistant to a nuclear incident (as defined in section 11 of the Atomic Energy Act of 1954 (42 U.S.C. 2014)); and"
- "lowers the cost of electricity over the licensed lifetime of an existing commercial nuclear reactor."

In coordination with the U.S. Department of Energy (DOE), several fuel vendors announced plans to develop and seek approval for ATF technologies by the mid-to-late 2020s, including the development of fuels featuring enhanced accident tolerance, higher burnup, and increased enrichment (NRC 2023-TN8675).

ATF technologies under development include coated zirconium-alloy (Zr-alloy) claddings, doped uranium dioxide (UO_2) pellets, iron-chromium-aluminum (FeCrAl) cladding, silicon carbide (SiC) cladding, uranium nitride (UN) pellets, and metallic fuels (NRC 2021-TN8017). The NRC staff anticipates that applicants, in addition to seeking to adopt ATF technologies, may also seek to use fuels with enrichments up to approximately 10 weight percent (wt%) uranium-235 (U-235) and higher burnup levels up to approximately 75 to 80 gigawatt days per metric ton of uranium (GWd/MTU). Current LWRs have enrichment levels of 3–5 wt% U-235 and reach burnup levels up to 62 GWd/MTU. Both enrichment and burnup increases would exceed the conditions of fuel

[1] This analysis in this document would apply to any subsequent NRC-licensed LWR using ATF technologies that feature increased enrichment and higher burnup levels.

and waste transport specified in Table S-4 in 10 CFR 51.52 (TN250). Based on these developments, the NRC staff considers the pursuit of increased enrichment and higher burnup a component of the ATF program.

Several NRC environmental reviews consider the environmental impacts of the uranium fuel cycle, including fuel fabrication, transport, and disposal of spent nuclear fuel (SNF) by incorporating the uranium fuel cycle environmental impact data in Table S-3 of 10 CFR 51.51 (TN250) by reference to bound the environmental impacts of the licensing action under consideration. The analysis of the environmental impacts for fuel compositions up to 5 wt% U-235 and up to 62 GWd/MTU are discussed in Section 4.12.1.1, Uranium Fuel Cycle, in Revision 1 of "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," also referred to as the 2013 version of License Renewal Generic Environmental Impact Statement (GEIS) and NUREG-1437 (NRC 2013-TN2654). The NRC staff considers the uranium fuel cycle environmental impacts data in Table S-3,[2] of 10 CFR 51.51 (TN250), bounding for all LWRs for fuels, as described in the 1996 and 2013 versions of the License Renewal GEIS (NRC 1996-TN288 and NRC 2013-TN2654, respectively).

The U.S. Atomic Energy Commission (AEC) and NRC staff assessed the environmental impacts of fuel and waste transportation in the "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," WASH-1238, published in December 1972, (AEC 1972-TN22), and "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, Supplement 1," NUREG-75/038, published in April 1975 (NRC 1975-TN216), which were then codified in Table S-4 of 10 CFR 51.52 (TN250). The analyses in WASH-1238 and NUREG-75/038 were based on 4 wt% U-235 enrichment and a 33 GWd/MTU average burnup level (AEC 1972-TN22; NRC 1975-TN216). Since then, the NRC staff has re-examined the risks of SNF transport and determined that the risks to the public are low for enrichment and burnup levels up to 5 wt% U-235 and 62 GWd/MTU burnup. The NRC staff concluded in the 2013 License Renewal GEIS that the values in Table S-4 of 10 CFR 51.52 (TN250) would still be bounding during the license renewal period, as long as (1) enrichment of unirradiated fuel was 5 wt% U-235 or less, (2) burnup of spent fuel was 62 GWd/MTU or less, and (3) higher burnup spent fuel (higher than 33 GWd/MTU) was cooled for at least 5 years before being shipped offsite (NRC 2013-TN2654).

In NUREG-0586, "Generic Environmental Impact Statement for Decommissioning Nuclear Facilities: Supplement 1, Regarding the Decommissioning of Nuclear Power Reactors," (Decommissioning GEIS) (NRC 2002-TN665), the NRC staff evaluated the environmental effects of decommissioning as residual radioactivity is reduced to levels that allow for the termination of the operating license. Based on its evaluation, the NRC staff determined generically that the impacts on certain environmental issues would be small, but that impacts for other environmental issues would be site-specific. Environmental issues with site-specific impacts need to be considered at the time of decommissioning. The study presented in this NUREG examined whether the environmental impacts analyzed in the Decommissioning GEIS would bound the deployment and use of ATF technologies in LWRs. If unbounded, impacts would need to be addressed in the environmental review for each application for use of ATF.

In addition, NUREG-1757, Volume 2, Revision 2, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," (NRC 2022-TN8031), provides guidance on compliance with the radiological criteria for LWR license termination in

[2] 10 CFR 51.51(b) (TN250): Table S-3—Table of Uranium Fuel Cycle Environmental Data.

10 CFR Part 20 (TN283), “Standards for Protection against Radiation,” Subpart E, “Radiological Criteria for License Termination.” The evaluations of the environmental effects of decommissioning in NUREG-1757 were based on current fuel characteristics of 5 wt% U-235 or less and burnup levels of 62 GWd/MTU or less.

In this study, the NRC staff is evaluating environmental documents and assessing available fuel performance analyses, data, and NRC-sponsored ATF studies with the goal of determining the generic environmental effects of deployment and use of ATF technologies in LWRs given increased enrichment and higher burnup. This study of the deployment and use of ATF with increased enrichment and higher burnup levels in LWRs addresses environmental issues associated with the uranium fuel cycle (10 CFR 51.51, Table S-3, Table of Uranium Fuel Cycle Environmental Data [TN250]), fuel and waste transportation (10 CFR 51.52, Table S-4, Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor [TN250]), and LWR decommissioning.

1.3 Scope of this Study

The NRC’s ATF Project Plan outlines staff efforts to prepare to review license applications related to the deployment and use of ATF technologies in LWRs as well as ongoing discussions concerning regulatory issues for in-reactor performance, fuel facilities, transportation, and storage (NRC 2021-TN8017). The ATF technologies would be subject to current uranium fuel cycle enrichment levels, transportation requirements based on enrichment and SNF burnup levels, and current LWR decommissioning assumptions.

To support efficient and effective licensing reviews of ATFs and to reduce the need for a complex site-specific environmental review for each ATF LAR, this study evaluated the reasonably foreseeable impacts of near-term ATF technologies. Industry has indicated its desire to also increase the enrichment of the uranium above 5 wt% U-235 and extend the burnup to levels above 62 GWd/MTU in current LWR fuels and ATF. Accordingly, the NRC staff also assesses these impacts in this study, with increased enrichment and higher burnup levels of greater than 5 wt% U-235 and greater than 62 GWd/MTU, respectively, on the uranium fuel cycle, transportation of fuel and waste, and decommissioning for LWRs (i.e., a bounding analysis). The NRC staff assessed and applied NRC-sponsored ATF technology reports; prior environmental reviews (such as the 2013 License Renewal GEIS [NRC 2013-TN2654] and Decommissioning GEIS [NRC 2002-TN7254]); transportation studies; and new or updated data sources to determine the bounding (generic) environmental impacts of ATF technologies with increased enrichment and higher burnup levels in LWRs.

A 60-day comment period began on September 1, 2023, when the NRC published a *Federal Register* Notice for the draft report for comment of NUREG-2266 (88 FR 60507-TN9593) to allow members of the public and other interested parties to comment on the study through Regulations.gov under Docket ID NRC-2023-0113. Two members of the public and two organizations provided comments on the draft NUREG-2266. Appendix F presents the comments received on the draft NUREG-2266, with responses to the comments, and indicates whether and where the draft NUREG-2266 was revised as a result of a comment.

1.4 Accident Tolerant Fuel Technologies Under Consideration in this Study

Three of the largest nuclear fuel suppliers in the United States (Westinghouse Electric Company [Westinghouse], Framatome Inc. Fuel Fabrication Facility [Framatome FFF], and Global Nuclear Fuels – Americas) are working with the DOE to develop ATF technologies for the nation’s fleet of LWRs (DOE 2022-TN8021). This evaluation addresses the environmental impacts associated with the near-term, first-generation ATF technologies, coated cladding and doped pellets, along with the near-term, second-generation ATF technology, FeCrAl cladding. This section also briefly describes longer-term ATF concepts under development. However, it is unclear at this time what the potential impacts from these longer-term technologies might be; thus, those impacts would have to be evaluated at an appropriate time in the future.

Fuel vendors assert that ATF technologies will provide better fuel performance during severe accident conditions and design basis accident conditions. While none of these ATF technologies has been approved for use beyond lead test assembly insertion, the NRC staff anticipates that the design features that provide improved behavior during accident conditions may allow the applicants to request approval to use higher burnup levels than those of traditional fuel. Applicants may request approval of higher U-235 enrichment levels to also enable higher burnup operation. All of these ATF technologies are emerging technologies and have experience with only the first cycle of lead test assemblies. Thus, some of the information is preliminary. The NRC staff will reassess the environmental impacts of ATF, including ATF with increased enrichment and higher burnup levels, if new information becomes available, such as during review of LARs.

To justify the use of ATF, fuel vendors must demonstrate compliance with existing specified acceptable fuel design limits (SAFDLs). One way of doing so is described in the NUREG-0800, "*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*," (NRC 2007-TN613) and is found in 10 CFR Part 50. Additionally, ATF may be subject to additional SAFDLs that have been identified to address damage mechanisms specific to these technologies (PNNL 2019-TN8288, PNNL 2020-TN8289). However, the NRC staff anticipates that ATF will be discharged from the reactor in the same or better condition than traditional fuel because licensees are required to perform a fuel system safety analysis. The NRC staff also anticipates that the current regulatory framework for SNF storage and transportation of near-term ATF (first- and second-generation) will be generally acceptable (PNNL 2020-TN8290).

For this study, the effects of near-term ATF will be evaluated with increased U-235 enrichment and higher burnup levels compared to those of current fuel designs. The fuel parameters that need to be discussed are presented in Appendices A and B, and include radionuclide source term, fraction of failed fuel rods, gap inventory, release of crud (i.e., buildup of corrosion products on surfaces), and release of particulates such as cesium (Cs) and rubidium (Rb) and noble gases such as xenon (Xe) and krypton (Kr) from failed fuel. Also, as noted in the study by Hall et al. (2021-TN8286), calculations of isotopic content changes associated with Cr-coated cladding and doped pellets, such as in radionuclide inventory, demonstrated negligible effects of ATF versus non-ATF for enrichments of 5 and 10 wt% U-235 and burnup of 62 and 80 GWd/MTU.

1.4.1 Coated Cladding

Historically, nuclear fuel for LWRs has usually consisted of Zr-alloy cladding with UO_2 fuel pellets. Nuclear fuel vendors, such as Framatome FFF, are currently conducting research and testing ATF with the outside of the Zr-alloy cladding coated with a thin layer of either chromium or a proprietary material. Nuclear fuel vendors claim these coatings would provide enhanced protection of ATF rods against debris fretting and oxidation resistance and superior material behavior over a range of conditions (NRC 2024-TN10184).

Based on the information available at this time, the use of coated cladding is not expected to alter the environmental impact of SNF. Any coated cladding approved for in-reactor use will be required to maintain an acceptable level of strength and ductility across the full spectrum of burnup to meet established SAFDLs (Geelhood et al. 2018-TN8677). In general, the strength and ductility are the properties that would preclude damage to the fuel during storage and transportation. The requirements for these properties are more stringent for in-reactor periods than for conditions of storage and transport (PNNL 2020-TN8290). Because the NRC staff does not expect the strength and ductility of coated cladding to be affected by the introduction of a thin coating, the cladding failure probability would not increase relative to that of standard cladding (PNNL 2019-TN8288). Thus, the use of coated cladding will not affect the fuel pellet source term beyond the burnup and enrichment effects. Likewise, the coated cladding does not affect the in-reactor pellet temperature (PNNL 2019-TN8288) and therefore will not affect the release or production of fission gas, which are both temperature-driven, and subsequent pellet release of particulates, such as Cs and Rb, or noble gases, such as Xe and Kr. The NRC staff anticipates that the use of Cr-coated cladding will result in lower instances of rod failure during transportation accidents because the Cr coating reduces in-reactor oxidation and hydrogen pickup, which are the primary in-reactor mechanisms that reduce the strength and ductility of fuel cladding. Regarding crud, there is no evidence that crud would preferentially accumulate on Cr-coated fuel rods. This is because crud formation is primarily controlled through coolant chemistry and to a lesser degree by surface roughness. The Cr coating will likely have the same final surface finish as traditional cladding.

Therefore, the environmental effects of coated cladding can be assessed based on the performance of traditional fuel with potentially higher burnup and U-235 enrichment.

1.4.2 Doped Pellets

For many years, nuclear fuel vendors have been conducting research and testing fuel pellets that mix other materials, known as dopants, into the pellets during the manufacturing process. These “doped” pellets have been approved for use in BWRs but approval of dopants for PWR applications is being developed as an ATF technology. These dopants slightly change the physical properties of the resulting fuel pellets by increasing the ceramic grain size. Nuclear fuel vendors claim that there are two advantages of doped pellets over existing designs. The first would be to produce a slightly softer pellet to reduce the risk of cladding damage due to pellet clad interaction during power maneuvers, and doing so has been approved and used in BWRs for many years. The second purported advantage is the increased ceramic grain size, which fuel vendors anticipated would promote fission gas retention within the fuel pellet, which may decrease the radioactive gases in the fuel-cladding gap. However, existing experience with doped pellets and large-grained pellets have indicated little to no impact of these features on the fission gas release from these pellets relative to standard pellets (Richmond and Geelhood 2018-TN8678). These doped pellets have recently been batch loaded by reactor licensees, such as in Brunswick Steam Electric Plant (Brunswick) Units 1 and 2 (NRC 2023-TN8023). In March 2023, the NRC-approved Westinghouse’s Advanced Doped Pellet Technology (ADOPT) fuel pellets for use in PWRs (Westinghouse 2022-TN8287).

Based on the information available at this time, the use of doped pellets is not expected to alter the environmental impact of SNF. The quantities and types of dopants being proposed for ATF designs are such that they will not affect the nuclear properties (fission rate and fission yield) of the pellets, and the existing source term calculations are expected to be representative of these pellets. The dopants often result in larger fuel grain size, but the overall fission gas release performance of the doped and undoped fuel is similar such that the gap inventories are expected to be the same (Richmond and Geelhood 2018-TN8678). Likewise, dopants will not affect the release or production of fission gas and subsequent pellet release of particulates, such as Cs and Rb, or noble gases, such as Xe and Kr. Fuel failure and crud buildup are driven by the cladding performance and are not affected by doped pellets.

Therefore, environmental effects of doped pellets can be assessed based on the performance of traditional fuel with potentially higher burnup and U-235 enrichment.

1.4.3 Iron-Chromium-Aluminum Cladding

FeCrAl cladding is a near-term, second-generation ATF technology under development by nuclear fuel vendors. As an alternative to Zr-alloys that have been used for fuel rod cladding for the past 40 years, an FeCrAl-based alloy is being developed by Oak Ridge National Laboratory (ORNL) in partnership with Global Nuclear Fuel – Americas (NRC 2023-TN8024). The possible advantages of FeCrAl cladding are improved high-temperature steam oxidation (lower equivalent cladding reacted and hydrogen generation under accident conditions), improved strength at normal operating conditions and high-temperature accident conditions, and improved normal operation corrosion performance. Licensees have inserted lead test assemblies containing FeCrAl cladding into LWRs to collect technical and performance data to support development of this ATF technology.

Based on the information available at this time, the use of FeCrAl cladding is not expected to alter the environmental impact of SNF. Any FeCrAl cladding approved for in-reactor use will be required to maintain an acceptable level of strength and ductility across the full spectrum of burnup to meet established SAFDLs (Geelhood et al. 2018-TN8677). In general, adequate strength and ductility are the properties that would preclude damage to the fuel during storage and transportation. The requirements for these properties are more stringent for in-reactor periods than for conditions of storage and transport (NRC 2007-TN613 [Chapter 4, Reactor, Section 4.2, Fuel System Design], PNNL 2020-TN8290). Although FeCrAl cladding will likely have a thinner wall than Zr-alloy cladding owing to a significant reactivity penalty from the iron (Hall et al. 2021-TN8286), the NRC staff expects the overall rod strength and ductility of FeCrAl to be the same or greater than Zr-alloy cladding because of the strengths and ductility requirements for in-reactor operation. Therefore, the cladding failure probability during spent fuel storage and transportation, which is driven by cladding strength and ductility, would not increase relative to that of Zr-alloy cladding. Hence, the use of FeCrAl cladding would not affect the fuel pellet source term beyond the burnup and enrichment effects.

The NRC staff does not expect FeCrAl cladding to result in adverse environmental impacts with respect to the presence of iron in the cladding. This will result in an overall increase of the cobalt-60 (Co-60) in the overall assembly source term. However, radionuclides in the cladding are not dispersible under transportation accident scenarios because the temperature of these events will not melt the cladding and therefore will not affect these transportation accident analyses. The use of FeCrAl cladding will not affect the release or production of fission gas, which are temperature-driven, and subsequent pellet release of particulates, such as Cs and Rb, or noble gases, such as Xe and Kr.

Regarding the fuel failure fraction, although FeCrAl cladding is expected to be thinner than traditional cladding, the strength of FeCrAl is greater than Zr-alloy cladding and will have the same or greater post-irradiation strength and ductility as Zr-alloy cladding. Hence, the use of FeCrAl cladding would result in the same or lower instances of rod failure during transportation accidents as with current fuel pins.

There is no evidence that crud would preferentially accumulate on FeCrAl fuel rods because crud formation is primarily controlled through coolant chemistry and to a lesser degree by surface roughness. The FeCrAl cladding will likely have the same final surface finish as traditional cladding. Given that testing of FeCrAl cladding is ongoing, additional performance data would be provided to clarify the above discussion if this ATF technology is to be deployed.

The NRC staff will confirm this analysis either in an update to its generic assessment or in the site-specific environmental review on an LAR to use FeCrAl. Therefore, given our current knowledge of FeCrAl cladding, the environmental effects of FeCrAl cladding can be assessed based on performance of Zr-alloy cladding with potentially higher U-235 enrichment and burnup levels.

1.4.4 Longer-Term Accident Tolerant Fuel Technologies

In addition to the near-term ATF technologies discussed above, the nuclear industry is also developing several longer-term ATF technologies, such as UN pellets, SiC cladding, and extruded metallic fuel (NRC 2023-TN8025). These technologies need additional research and development, and implementation may be many years into the future. Research into the replacement of UO_2 with UN in fuel pellets to promote higher power levels, longer nuclear fuel cycles, high melting points, improved neutronic performance, and enhanced thermal conductivity to promote lower operating temperatures is ongoing. Nuclear fuel vendors are developing several SiC composite cladding materials where SiC fibers are woven, then impregnated with additional SiC to form a rigid tube. The potential benefits of SiC cladding are to maintain structural integrity at very high temperatures and improve high-temperature steam oxidation for longer accident coping times and less hydrogen generation under design basis accident and severe accident conditions. Extruded metallic fuel is a new fuel design that incorporates an extruded metallic bar composed of a zirconium-uranium matrix within a Zr-alloy cladding. The potential benefits of extruded metallic fuel are a significant increase in fuel thermal conductivity, complete retention of fission products, and support of higher power and longer fuel cycles.

These longer-term ATF technologies are still under development, and it is not possible to evaluate the impact of their use on the environmental effects of storage and transportation of SNF. Therefore, an assessment of these technologies is outside the scope of this report. Once longer-term ATF technologies are more fully developed, their environmental impacts would be revisited to determine whether or not they fit within the analysis of this study.

1.5 Organization of the Study

The evaluation presented in this study examines the environmental implications of deployment and use of ATF technologies in LWRs: Section 1 is the introduction; Section 2 discusses the environmental effects of changes to the front and back segments of the uranium fuel cycle related to ATF technologies, including continued storage after the cessation of operations; Section 3 describes and analyzes the environmental effects of transportation of unirradiated ATF and waste to and from LWRs; Section 4 examines the environmental implications of the deployment and use of ATF technologies for decommissioning activities in LWRs; and Section 5 provides conclusions. Appendices are provided for input parameters and technical information necessary to support the transportation analysis including sensitivity calculations.

2 URANIUM FUEL CYCLE

2.1 Introduction

2.1.1 Uranium Fuel Cycle Environmental Data

As discussed in Section 3.12.1.1, Uranium Fuel Cycle, of the 2013 License Renewal GEIS, the NRC evaluated the environmental impacts that would be associated with operating uranium fuel cycle facilities other than reactors in two NRC documents: WASH-1248 (AEC 1974-TN23) and NUREG-0116 (NRC 1976-TN292). The types of facilities and their environmental impacts considered in these two documents include the following:

- uranium mining – facilities in which the uranium ore is mined;
- uranium milling – facilities in which the uranium ore is refined to produce uranium concentrates in the form of triuranium octaoxide (U_3O_8);
- uranium hexafluoride (UF_6) production – facilities in which the uranium concentrates are converted to UF_6;
- isotopic enrichment – facilities in which the isotopic ratio of the U-235 isotope in natural uranium is increased to meet the requirements of LWRs;
- fuel fabrication – facilities in which the enriched UF_6 is converted to UO_2 and made into sintered UO_2 pellets. These facilities also encapsulate the pellets in fuel rods and assemble the rods into fuel assemblies ready to be inserted into reactors;
- reprocessing – facilities that disassemble the spent fuel assemblies, chop up the fuel rods into small sections, chemically dissolve the spent fuel out of sectioned fuel rod pieces, and chemically separate the uranium in spent fuel from the plutonium for reuse and from other radionuclides (primarily fission products and actinides); and
- disposal – facilities that would bury radioactive wastes. Radioactive waste can be designated as either low-level radioactive waste (LLRW) or high-level radioactive waste (HLW). The NRC staff anticipates that HLW would be disposed of in a deep geologic repository that would accept, among other things, SNF that is removed from the reactors and not reprocessed as well as certain wastes from reprocessing of spent fuel. The LLRW is disposed of in near-surface disposal facilities. All fuel cycle facilities generate at least small amounts of LLRW during operations.

In addition to evaluating the environmental impacts occurring at the above facilities, WASH-1248 and NUREG-0116 evaluated the environmental impacts associated with the transportation of radioactive materials among these facilities (e.g., enriched uranium from the isotopic enrichment facility to the fuel fabrication facility). The analysis in WASH-1248 is based on the principal environmental considerations for each component of the uranium fuel cycle, and the aggregate considerations, normalized to the annual fuel requirement of a 1,000 megawatt-electric (MWe) (3,000 megawatt-thermal [MWt]) model LWR (AEC 1974-TN23). This normalization is called the “annual model LWR fuel requirement” throughout WASH-1248 (AEC 1974-TN23). The NRC summarized the results of this analysis in a table promulgated as Table S-3 in 10 CFR 51.51(b) (TN250).

Figure 2-1 is a schematic representation of the uranium fuel cycle for an LWR. It shows the major uranium flows and major uranium processing facilities. It also shows reprocessing with the production and use of mixed-oxide fuel. The operations in the later stages, reprocessing, and production and use of mixed-oxide fuel are not currently planned for ATF for LWRs. However, this could change at a future time. Table S-3 addresses environmental impacts related to the uranium fuel cycle but does not address mixed-oxide fuel or advanced nuclear reactor fuels produced through reprocessing. The assumption applied for Table S-3 regarding plutonium recovered from recycling was that the recovered plutonium would be placed into storage for future use (see Figure S1 of WASH-1248 [AEC 1974-TN23]).

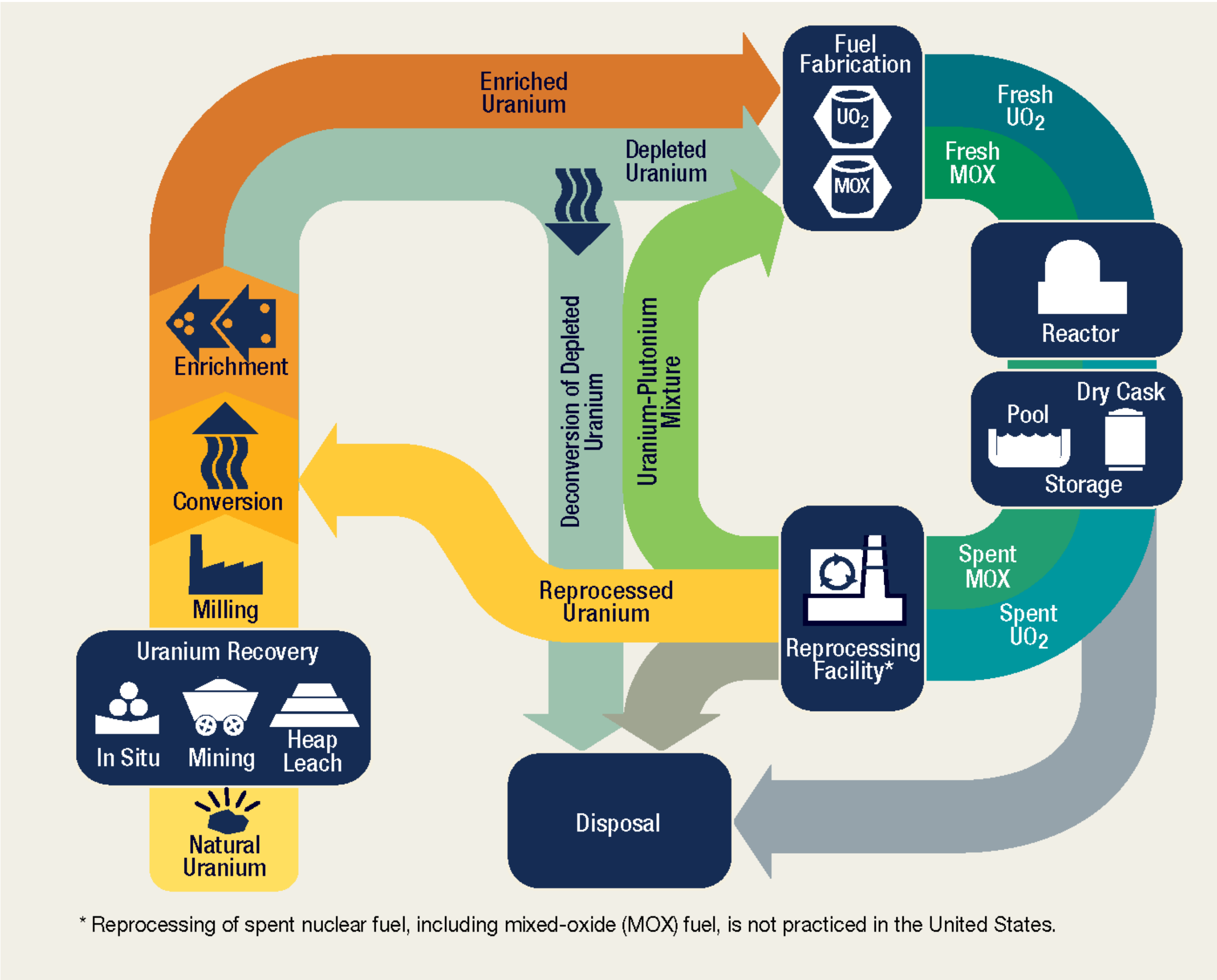

Figure 2-1 Options of the Current Fuel Cycle, Which Includes the Table S-3 Uranium Fuel Cycle (NRC 2019-TN6652)

The 1996 version of the License Renewal GEIS (NRC 1996-TN288), applying Table S-3, found the environmental effects of the once-through (i.e., no reprocessing), low enriched uranium (LEU)[1] fuel cycle to be small.[2] The NRC codified these findings in 10 CFR Part 51 (TN250), Appendix B and Table B-1, Summary of Findings on NEPA Issues for License Renewal of

[1] As defined in 10 CFR 50.2 (TN249), LEU fuel means fuel in which the wt% of U-235 in the uranium is less than 20 percent.

[2] Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.

Nuclear Power Plants. The NRC updated that environmental impacts determination in the 2013 License Renewal GEIS (NRC 2013-TN2654). In Section 4.12.1.1 of the 2013 License Renewal GEIS, the NRC staff reassessed the environmental effects listed in Table S-3 and concluded that no new information had been identified that would alter the conclusion in the 1996 version of the License Renewal GEIS. The analyses provided in Section 4.12.1.1 to the 2013 License Renewal GEIS are incorporated by reference into this analysis and support the following evaluation.

2.1.2 Changes in the Uranium Fuel Cycle Since WASH-1248

Many of the uranium fuel cycle facilities and processes assessed for their environmental effects and inclusion in Table S-3 still exist today. However, some have undergone several industrial improvements and technological advances that have significantly reduced their environmental effects. As discussed in the 1996 and 2013 versions of the License Renewal GEIS (NRC 1996-TN288, NRC 2013-TN2654), the uranium fuel cycle facilities changes since Table S-3 was originally prepared include increased enrichment up to 5 wt% U-235 and higher burnup levels up to 62 GWd/MTU. The NRC staff concluded that even though certain fuel cycle operations and fuel management practices have changed over the years, the basis and methodology used in preparing Table S-3 are conservative enough that the impacts described by the use of Table S-3 are still bounding. For the reasons discussed below, the NRC staff determined that this conclusion still holds for traditional fuel (NRC 2013-TN2654).

The above conclusion that Table S-3 would still be bounding for traditional fuel is based on the following recent uranium fuel cycle trends in the United States:

- Increasing use of in situ leach uranium mining, which does not produce mill tailings and would lower the release of radon gas. A discussion of this subject is provided in Section 2.2.1.
- Transitioning of U.S. uranium enrichment technology from gaseous diffusion to gas centrifugation. The latter process uses only a fraction of the electrical energy per separation unit compared to gaseous diffusion. This topic is discussed in Section 2.2.2.
- Electricity sources to support all fuel cycle facility operations are less dependent on electricity derived from the burning of coal than was assumed in the WASH-1248 analysis for the impacts codified in Table S-3 as discussed later in this section.
- Current LWRs are using nuclear fuel more efficiently because of higher levels of fuel burnup. Thus, less uranium fuel per year of reactor operation is required now than in the past to generate the same amount of electricity (an increase in the time for refueling from 12 months to 18 months or greater).

The Table S-3 values were calculated from industry-based experience in the domain of the performance of each type of facility or operation within the fuel cycle. Recognizing that this approach meant that there would be a range of reasonable values for each estimate, the NRC staff chose assumptions or factors so that the calculated environmental impact would not be underestimated. The NRC staff intended for this approach to make sure that the actual environmental impacts would be less than the quantities shown in Table S-3 for all LWR NPPs within the widest range of operating conditions. The NRC staff recognizes that many of the fuel cycle parameters and interactions vary in small ways from the estimates in Table S-3 and concludes that these variations would have negligible impacts on the Table S-3 calculations. For example, to determine the quantity of fuel required for a year’s operation of a NPP in Table S-3, the NRC staff defined the reference reactor as a 1,000 MWe LWR operating at 80 percent capacity with a 12-month fuel-reloading cycle and an average fuel burnup of 33 GWd/MTU. The current LWR fleet is operating with an average operating capacity factor of approximately 95 percent with peak fuel rod burnup of up to 62 GWd/MTU and with refueling occurring at some LWRs at intervals as great as approximately 2 years (NRC 2018-TN6254, NRC 2019-TN6136).

The original Table S-3 analysis from the 1970s was developed when most of the electricity generated in the United States was produced in plants that burned fossil fuels with coal comprising the bulk of fossil-fuel utilization (AEC 1974-TN23). However, today the energy sources for utility-scale electrical generation are very diverse with (DOE/EIA 2023-TN8285):

- 19.5 percent from coal and this percentage is decreasing
- 39.8 percent from natural gas, for which air emissions are much less than those from coal
- 18.2 percent from NPPs
- 21.5 percent from renewables and is increasing (15.3 percent from non-hydroelectric renewables and 6.2 percent from hydroelectric)
- less than 1 percent from petroleum and other sources

The use of coal for producing electricity results in the production of significantly more air emissions and liquid pollutants than emitted by other sources of electricity that are more prevalent today. Consequently, the significant increase in electricity production from nuclear, natural gas, and renewables compared to coal means that the environmental effects of the production of electricity necessary for the uranium fuel cycle are less than those assessed in WASH-1248. Thus, the various environmental data provided in Table S-3 related to air emissions, liquid pollutants, and water/thermal values characterize impacts that clearly exceed those from today's electrical generation contribution to the uranium fuel cycle. Therefore, the NRC staff has determined the current environmental impacts from the uranium fuel cycle would be bounded by the coal-electrical generation data assessed by WASH-1248 (AEC 1974-TN23) and codified in Table S-3. This trend of decreasing reliance on fossil fuels for electrical generation will continue, spurred by actions to combat climate change[3] (DOE/EIA 2020-TN6653).

Based on several of these factors, the 2013 License Renewal GEIS states:

> It was concluded that even though certain fuel cycle operations and fuel management practices have changed over the years, the assumptions and methodology used in preparing Table S-3 were conservative enough that the impacts described by the use of Table S-3 would still be bounding.

With Table S-3 values being still bounding for the LWR uranium fuel cycle, the following sections provide a brief background about the components of the uranium fuel cycle and discuss how deployment and use of ATF, including ATF technologies with increased enrichment

[3] The NRC defines Climate Changes as, "The Changes in the Earth's surface temperature thought to be caused by the greenhouse effect and responsible for changes in global climate patterns. The greenhouse effect is the trapping and buildup of heat in the atmosphere (troposphere) near the Earth's surface. Some of the heat flowing back toward space from the Earth's surface is absorbed by water vapor, carbon dioxide, ozone, and certain other gases in the atmosphere and then reradiated back toward the Earth's surface." (NRC 2014-TN4117)

and higher burnup, would affect the uranium fuel cycle with respect to the impacts presented in Table S-3.

2.2 Uranium Fuel Cycle Impacts Due to Accident Tolerant Fuel Deployment

The NRC evaluates uranium fuel cycle impacts of the reactor fuels to meet its obligations under NEPA, as amended (TN661). The NRC has generically evaluated the environmental effects of the uranium fuel cycle for LWRs that use Zr-alloy-clad, UO_2 fuel. The results of the evaluation are presented in 10 CFR 51.51(b) (TN250), Table S-3, Table of Uranium Fuel Cycle Environmental Data. While 10 CFR 51.51 (TN250) specifically addresses LWRs being licensed at the construction permit stage, early site permit stage, or combined license stage, uranium fuel cycle changes in support of the deployment and use of ATF are a connected action under NEPA (40 CFR § 1501.9(e)(1)), requiring an appropriate NRC staff evaluation. The deployment and use of ATF would require changes to specific segments of the uranium fuel cycle if, for example, there were increases in enrichment percentages with accompanied higher burnup levels. As discussed below, the environmental impacts caused by uranium recovery and conversion from the deployment and use of ATF would be less than those described in the discussion in 2013 License Renewal GEIS for these segments of the uranium fuel cycle. The deployment and use of ATF with increased enrichment and higher burnup levels could affect all portions of the uranium fuel cycle. Additionally, this section considers the effect of higher burnup levels with respect to the analysis in the Continued Storage GEIS, NUREG-2157 (NRC 2014-TN4117). As a final note, fuel only has a higher burnup after it has been used in a reactor. As such, when considering the environmental effects of higher burnup, this difference is only relevant on the back-end of the fuel cycle. Thus, sections discussing the front-end of the fuel cycle do not discuss differences caused by higher burnup.

2.2.1 Uranium Recovery and Conversion

The analyses for Table S-3 regarding uranium recovery were predicated on active uranium mining, heap leaching, and large industrial milling facilities (AEC 1974-TN23). There were no active traditional uranium mines (i.e., shallow open pits or underground) and active heap leaching sites in the United States during 2022 (DOE/EIA 2023-TN8065). The technology applied today to extract natural uranium from the ground has changed significantly since the publication of WASH-1248, namely the use of in-situ recovery that avoids many of the adverse environmental impacts where uranium ore is removed from deep underground shafts or shallow open pits. In May 2009, the NRC staff published the “Generic Environmental Impact Statement for In-Situ Leach Uranium Milling Facilities” (NUREG-1910, NRC 2009-TN2559), which addresses common environmental issues associated with the building, operating, and decommissioning of facilities, as well as the groundwater restoration at such in-situ recovery facilities. As discussed in NUREG-1910, the in-situ recovery process does not involve removal of large volumes of uranium ore from a site; transport of the uranium ore to a large milling facility; and processing of the uranium ore resulting in tailing piles and leachate ponds with potential environmental impacts due to chemical contamination of water sources and the associated release of radon gas. Therefore, the environmental impacts for in-situ recovery are less than those listed in Table S-3 for uranium recovery facilities.

The effect of ATF deployment and use on uranium recovery, by itself, would not change the level of impacts described in NUREG-1910. However, increasing enrichment would require an increase in natural uranium feedstock on the front-end of the uranium fuel cycle. For example, by approximately doubling the uranium feedstock from 4 wt% U-235 of WASH-1248 to 7 wt% U-235, the back-end would be benefited because it would reduce the energy-normalized quantity of spent fuel waste (Burns et al. 2020-TN8026). While the quantity of metric tons (MT) of uranium yellowcake, a form of natural uranium oxide as U_3O_8 from uranium recovery needed on an annual basis would increase over current annual uranium supply quantities, the reduced environmental impacts of in-situ recovery would offset the increased impacts of the need for uranium (see Section 2.2.1 of NRC 2016-TN5487), and there is an adequate supply of yellowcake from domestic (DOE/EIA 2022-TN8027) or foreign sources. Thus, this would result in a lessening of the environmental impacts as described in NUREG-1910 on a per reactor basis. Since Table S-3 bounds the uranium recovery impacts in NUREG-1910, Table S-3 would continue to bound the environmental effects of uranium recovery impacts from ATF deployment and use including those resulting from increased enrichment.

With regard to uranium conversion, one U.S. uranium conversion facility, the Metropolis Works Plant (MTW) in Massac County, Illinois, uses a “dry”, or hydrofluorination process with gaseous reagents in fluidized bed reactors and distillation columns (NRC 2019-TN6964). Another uranium conversion process applies a “wet” process that starts with dissolving the yellow cake in nitric acid and purifying it by solvent extraction. As noted in Section 6.2.3 of the 1996 License Renewal GEIS (NRC 1996-TN288), in both process cases the environmental releases are so small that changing from 100 percent use of one process to 100 percent use of the other would make no significant difference in the total effects analyzed in WASH-1248.

ATF deployment and use with increased enrichment levels would result a greater amount of yellowcake to be processed during uranium conversion to UF_6 to support increased enrichments. By applying the UxC Fuel Cost Calculator (UxC 2023-TN8086), increasing enrichment to 8 wt% U-235 would need approximately 2.1 times more yellowcake feedstock than the 4 wt% U-235 that underscores Table S-3 environmental data. Increasing enrichment to 10 wt% U-235 would require approximately 2.6 times more yellowcake for UF_6 conversion than for 4 wt% U-235.

To assess the resulting uranium conversion environmental impacts for the increase in the amount of yellowcake for increased enrichments, the NRC staff first compared the environmental data provided in Table S-3 (10 CFR 51.51(b) [TN250]) to the uranium conversion environmental data provided in Table C-1 of WASH-1248 (AEC 1974-TN23). The uranium conversion environmental data in Table C-1 is based on a capacity of approximately 5,000 MT, which results in a small fraction of the total natural resource values provided in Table S-3. For a number of environmental considerations in Table S-3, a larger amount of yellowcake processed through uranium conversion would not cause a significant change in land, water, electricity, and most nonradiological and radiological gaseous and liquid effluent releases. The environmental data for which uranium conversion could have a more significant contribution are those related to natural gas, fluoride effluent releases, and liquid nonradiological effluent releases.

In 2019, the NRC published a final environmental assessment (EA) for the license renewal of the MTW UF_6 conversion facility, concluding with a finding of no significant impact based on an annual uranium conversion capacity of 15,000 MT and the highest production at about 13,000 MT (NRC 2019-TN6964). In making this environmental determination, the NRC staff evaluated all areas of environmental considerations. For example, Table 2-3 of the final EA lists the nonradiological air emissions from this facility over a period of 5 years (i.e., 2010 to 2014), which are in regulatory compliance. As for natural gas, the principal environmental effect of MTW’s operation is the release of greenhouse gases where the MTW released a small percentage, approximately 0.008 percent, of the estimated carbon dioxide generated in the State of Illinois. The final EA also documents for liquid nonradiological effluent releases, such as fluorides, that the MTW is operating in accordance with its National Pollutant Discharge Elimination System (NPDES) permit.

Therefore, given the three times larger capacity of the MTW than analyzed in WASH-1248 and the fact that the NRC staff found no significant impacts with that larger capacity for the MTW, the increase of approximately 2.6 times the amount of yellowcake conversion impacts contributing to the data in Table S-3 for increased enrichment of 10 wt% U-235 would not be significant. Hence, WASH-1248 and Table S-3 would still bound the environmental effects from the conversion of yellowcake to UF_6 for the deployment and use of ATF for 10 wt% U-235.

2.2.2 Uranium Enrichment

When considering the enrichment portion of the uranium fuel cycle the only relevant difference between traditional fuel and ATF with increased enrichment and higher burnup levels is the increased enrichment. During the enrichment process, uranium does not need to be treated differently if it will be used in an ATF or traditional fuel.

The uranium enrichment process has undergone significant changes since the analysis of Table S-3 provided in WASH-1248 (AEC 1974-TN23) and NUREG-0116 (NRC 1976-TN292). That analysis was based on gaseous diffusion enrichment, which had large energy requirements and, as discussed above, was primarily produced by coal-electrical generation plants that featured large air emissions and other environmental impacts, as noted in Table S-3.

Gaseous diffusion enrichment technology has been replaced by centrifuge enrichment technology, which requires significantly less energy to enrich uranium to similar or greater levels. This can be seen by comparing the work and energy necessary to produce 4 wt% and 10 wt% U-235. The separative work unit (SWU) is the standard measure of the work expended to separate isotopes of uranium (U-235 and uranium-238 [U-238]) during an enrichment process and is independent of the enrichment process (gaseous or centrifuge method). For the purposes of comparing the energy necessary to produce enriched uranium with either gaseous diffusion or centrifugation, the NRC staff determined the difference in energy usage between the two enrichment technologies, applying a unit mass of 1,000 kilograms [kg] (1 MT) of enriched uranium with enrichment tails assay of 0.25 wt% U-235 using the methodology of Napier (2020-TN6443)[4] with information from WASH-1248.

Using a SWU calculator (UxC 2023-TN8086) to obtain 1 MT of 4 wt% U-235, assuming a related amount of natural uranium, requires 5,832 SWUs for 0.25 wt% of U-235 in the tails. To obtain 1 MT of 10 wt% U-235 (high assay LEU) requires approximately 20,790 SWUs. The gaseous diffusion process consumes about 2,500 kilowatt-hours (kWh) per SWU, while modern gas centrifuge plants require only about 50 kWh per SWU (WNA 2020-TN6661). Thus, a centrifuge enrichment facility would consume approximately 1,040,000 kWh to reach 10 wt% U-235. A gaseous diffusion plant would consume approximately 51,975,000 kWh to produce the same amount of 10 wt% U-235. In fact, producing the same amount of 4 wt% U-235 by gaseous diffusion, which WASH-1248 and Table S-3 originally considered, requires approximately 14,600,000 kWh. Thus, a gaseous diffusion plant requires far more than the energy necessary to produce a similar amount of uranium enriched to 10 wt% U-235 with centrifuge enrichment.

[4] The NRC staff notes that the Napier report primarily describes the uranium fuel cycle for non-LWRs. Although, ATF is an LWR fuel, the NRC staff is only relying on the Napier report for the above SWU methodology and calculations, which are independent of reactor type, making the Napier report applicable in this limited circumstance.

On average, centrifuge enrichment uses approximately 104,000 kWh to increase enrichment by 1 percent (1,040,000 kWh divided by 10 wt% U-235) and gaseous diffusion uses approximately 3,650,000 kWh per 1 percent enrichment (14,600,000 divided by 4 wt% U-235). Hence, centrifuge enrichment uses about 97 percent less energy to enrich on a per percent basis. Since centrifuges are significantly more efficient for the enrichment of uranium over gaseous diffusion, Table S-3 would bound the environmental impacts from a centrifuge enrichment facility to produce the increased enrichment uranium expected for use in ATF assemblies.

2.2.3 Uranium Fuel Fabrication

Fuel fabrication facilities will need to be licensed to produce the necessary ATF types. The NRC currently regulates several different types of uranium fuel fabrication operations. For commercial NPP fuel, the following three fuel fabrication plants currently hold NRC licenses for processing LEU (NRC 2020-TN6835):

- Global Nuclear Fuel-Americas in Wilmington, North Carolina
- Westinghouse Columbia Fuel Fabrication Facility in Columbia, South Carolina
- Framatome FFF in Richland, Washington

The NRC also has licensed two other fuel fabrication plants to produce nuclear fuel for the U.S. Navy and to down blend highly enriched uranium with other uranium to create LEU reactor fuel for commercial NPPs. These two NRC-licensed fuel fabrication plants are the Nuclear Fuel Services plant in Erwin, Tennessee, and the BWXT Nuclear Operations Group plant in Lynchburg, Virginia (NRC 2020-TN8071). All five of the above-mentioned fuel fabrication facilities were in operation generating LWR fuel at the time of the WASH-1248 study, along with five other fuel fabrication facilities (AEC 1974-TN23).

In Appendix E of WASH-1248 (AEC 1974-TN23), a model fuel fabrication plant that had a capacity of 3 MTU/day and operated 300 days per year was used to assess environmental impacts for a total of 900 MTU/yr. WASH-1248 also assumed that the electricity used in fuel fabrication facilities came from coal power plants, some natural gas was used for process heat, and other external resources involved land use and water (AEC 1974-TN23). Since the publication of WASH-1248, a significant portion of the electricity produced by burning coal has been replaced by other cleaner electrical sources (DOE/EIA 2023-TN8285). At the time of WASH-1248, low enriched fuel fabrication facilities used a wet conversion process method for UF_6 to UO_2 conversion, which involves the use of ammonium hydroxide to form an intermediate ammonium diuranate (ADU) compound prior to final conversion to UO_2. Since WASH-1248, several of the above-mentioned fuel fabrication facilities now apply a dry process with less waste management environmental impacts than the ADU process. Only the Westinghouse Columbia Fuel Fabrication Facility currently applies the ADU process for final conversion to commercial nuclear fuel (NRC 2019-TN6472). As noted in Section 6.2.3 of the 1996 License Renewal GEIS (NRC 1996-TN288), this change from a wet to dry uranium conversion process reduces environmental impacts, but the impacts from uranium conversion are so small that the changes are not significant.

The deployment and use of near-term ATF technologies would not significantly change the processes at the various fuel fabrication facilities since the only significant change is the increased enrichment level and not the chemical form of the fuel. With regard to coated cladding and FeCrAl, the ATF cladding would be included as supplied material entering a fuel fabrication facility. The fuel fabrication facility would then use that cladding instead of the traditional zirconium-alloy cladding. With regard to doped pellets, the fuel fabrication facility would mix a chemical powder into the uranium oxide powder for doped uranium oxide pellets. All the other fuel fabrication processing steps would remain the same. Because the doped pellet technology is exchanging one material with another, applying these ATF technologies would not add any new process steps that would result in increases in existing effluent release streams. The effects of increased enrichment on fuel fabrication principally affects the criticality safety program and does not introduce any new or additional environmental impacts. Since fuel fabrication of ATF would have the same or similar impacts as traditional fuel fabrication, Table S-3 is still bounding for ATF technologies during fuel fabrication.

2.2.4 Reprocessing

As of the date of publication of this NUREG, there are no licensing actions before the NRC for the reprocessing of SNF from LWRs. In 2021, the NRC staff issued SECY-21-0026, which provided the NRC staff's assessment that a continued rulemaking effort on that subject was not then justified. The Commission approved the staff's recommendation and directed the NRC staff to continue to interact with DOE, international counterparts, and the industry to monitor activities related to an interest in reprocessing, including the licensee's application for reprocessing for advanced reactors, and to engage the Commission as appropriate (NRC 2022-TN8028). Some interest has been expressed and more is expected from potential applicants for reprocessing facilities, including advanced reactor designers, in the near-term use of reprocessed spent fuel (DOE 2022-TN8066).

Because deployment and use of ATF results in longer 24 month refueling times compared to the 12 months assumed for the analysis in WASH-1248 and NUREG-0116, there would be a reduction in the number of ATF assemblies available for reprocessing than with existing LWR fuel. Additionally, if reprocessing is pursued in the future, the industrial process to be implemented could be significantly different with fewer environmental impacts than those analyzed in Appendix F of WASH-1248. Given that industry does not currently reprocess spent fuel as part of the uranium fuel cycle, the NRC staff does not need to reach a conclusion about the impacts the deployment and use of ATF would have with regard to reprocessing. Before reprocessing becomes part of the fuel cycle, the NRC staff would account for the environmental effects of reprocessing.

2.2.5 Storage and Disposal of Radiological Wastes

Appendix G of WASH-1248 presents an analysis of the environmental impacts of managing radioactive wastes from the uranium fuel cycle activities (AEC 1974-TN23). The analysis is for radioactive wastes that can be categorized as HLW and other than high-level, or LLRW. The HLW generally consists of certain wastes from reprocessing of spent fuel as well as SNF that are removed from the reactors and not reprocessed. These wastes contain fission products that are either contained in the spent fuel or separated from fissile material recovered from irradiated fuel during reprocessing. HLW is to be disposed of in a deep geologic repository. The LLRW result from operations involving UF_6 production, fuel fabrication, and fuel reprocessing. LLRW generally include all wastes, regardless of concentration or specific activity, that are not designated as HLW and will be disposed of in a near-surface LLRW disposal facility.

While WASH-1248 states the LLRW, which is generated during fuel cycle operations, is variable and difficult to estimate, the total LLRW volume generated during fuel cycle operations annually is estimated to be approximately 14,000 cubic feet [ft^3] (396 cubic meter [m^3]) for the model LWR considered by WASH-1248 (AEC 1974-TN23). This analysis also assumes that, with no further compaction of the waste, the final volume of packages containing the waste could be estimated to be approximately 20,000 ft^3 (566 m^3) per annual model LWR fuel requirement (14,000 ft^3 of waste and 6,000 ft^3 of packaging material). The 20,000 ft^3 is a fraction of the annual LLRW from all U.S. sources shipped to the four Agreement State-licensed LLRW disposal facilities (NRC 2013-TN2654). Therefore, the LLRW generated during fuel cycle operations can be disposed at the currently operating facilities. Additionally, Table 3.11.1 in the 2013 License Renewal GEIS shows that the actual volume of LLRW shipped offsite for 10 NPPs in 2006 was generally far less than that presented in WASH-1248.

Section 3.11.1.2 of the 2013 License Renewal GEIS addresses the management of SNF at the existing NPPs where SNF is currently stored either in spent fuel pools or in Independent Spent Fuel Storage Installations (ISFSIs) using dry storage. When spent fuel is removed from a reactor, the fuel assembly is stored in racks placed in a spent fuel pool to isolate it from the environment and to allow the fuel rods within the fuel assembly to cool. When spent fuel pools are near capacity, utilities have sought other means of continued onsite storage. These include (1) expanded pool storage, (2) dry storage, (3) longer fuel burnup to reduce the amount of spent fuel requiring interim storage, and (4) shipment of spent fuel to other plants (NRC 2013-TN2654). Dry storage involves moving spent fuel assemblies that have been stored in the spent fuel pool for a certain period of time to shielded NRC-certified dry storage systems that are air cooled. The Commission concluded in both the 1996 and 2013 License Renewal GEISs that storage of existing spent fuel and storage of spent fuel generated during the licensing term can be accomplished safely and without significant environmental impacts during the license renewal period of the reactor, because radiation doses would be well within regulatory limits (NRC 2013-TN2654).

The analysis in WASH-1248 (AEC 1974-TN23) was based on 12-month refueling cycles, lower enrichment and burnup levels than are currently utilized for the current fleet of LWRs, along with the use of spent fuel pools exclusively for spent fuel storage. The higher burnup levels achieved since issuance of WASH-1248 result in greater utilization of the uranium fuel (i.e., greater efficiency in extracting energy from the fuel). This also has resulted in extended time between refueling operations and the removal of fewer fuel assemblies on a per reactor-year basis for many of the operating NPPs. Deployment and use of ATF with increased enrichment and higher burnup levels would result in further increases in fuel efficiency in extracting energy resulting in further reductions in the numbers of SNF assemblies removed during refueling operations for the same reasons (e.g., further extended time between refueling operations). With a reduced discharge rate of SNF from the deployment and use of ATF, the prior analysis of 1996 and 2013 License Renewal IGEIS would still apply (NRC 1996-TN288, NRC 2013-TN2654).

Recognizing that a HLW disposal facility, in which SNF would be disposed, did not yet exist, WASH-1248 stated that the AEC was proceeding on a program to design, construct, and operate a surface (or near-surface) facility in which the solidified commercial HLW would be stored in sealed canisters (AEC 1974-TN23). However, this program was never completed. Rather, in the late 1970s, the NRC examined an underlying assumption used in licensing reactors up to that time, namely that a repository could be secured for the ultimate disposal of spent fuel generated by nuclear reactors, and that spent fuel could be safely stored in the interim (NRC 2014-TN4117). On August 31, 1984, the Commission published the Waste Confidence decision (49 FR 34658-TN3370) and a final rule (49 FR 34688 1984-TN8030) that were codified into NRC regulations under 10 CFR 51.23 (TN250), “Temporary storage of spent fuel after cessation of reactor operation – Generic determination of no significant environmental impact.” The Waste Confidence decision was later revised to the Continued Storage Final Rule (79 FR 56238-TN4104). In particular, the Commission stated in the Continued Storage rulemaking that the environmental impacts of continued storage of SNF beyond the licensed life for operation of a reactor are those impacts identified in NUREG–2157 (79 FR 56249), and the NRC concluded that spent fuel can be safely managed in spent fuel pools in the short-term timeframe and dry casks during the short-term, long-term, and indefinite timeframes in the Continued Storage GEIS (79 FR 56253).

2.2.5.1 Evaluation of Continued Storage

Under 10 CFR 51.23(a) (TN250),

> [t]he Commission has generically determined that the environmental impacts of continued storage of SNF beyond the licensed life for operation of a reactor are those impacts identified in NUREG-2157, 'Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel.'

As stated in the Continued Storage GEIS (Volume 1, page 2-6. NRC 2014-TN4117), this generic analysis was "focused on past, present, and future spent fuel types that will be subject a future NRC licensing action." In particular, the analysis included commercial LWR fuel. The Commission evaluated the environmental impacts of continued storage of spent fuel that includes ATF. The information provided below is intended to provide a context and summary for the generic determinations made in the Continued Storage GEIS to aid the reader and is not intended to contradict nor reinterpret the information or determinations in the Continued Storage GEIS.

The complete history of the Waste Confidence decision, which has been referred to as Continued Storage since 2014, is provided in Section 1.1, History of Waste Confidence, of NUREG-2157, "Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel" (NRC 2014-TN4117) and is incorporated by reference. As a result of uncertainties regarding the timing of an operational geologic repository for a permanent disposal of SNF, the NRC developed and published the Continued Storage GEIS and revised 10 CFR 51.23 (TN250), which became "Environmental impacts of continued storage of SNF beyond the licensed life for operation of a reactor" (79 FR 56238-TN4104).

NUREG-2157, the Continued Storage GEIS, analyzes the environmental impacts of continued storage of spent fuel (NRC 2014-TN4117). In it, the NRC analyzed the direct, indirect, and cumulative effects of continued storage for the following three timeframes:

- short-term – 60 years beyond licensed life for reactor operations
- long-term – 100 years beyond the short-term storage timeframe
- indefinite – indefinite storage and handling of spent fuel

These timeframes are discussed in more detail in Section 1.8.2 of the Continued Storage GEIS (NRC 2014-TN4117). The locations of the storage sites related to these impacts were assessed for at-reactor storage, away-from-reactor storage, and cumulative impacts when added to other past, present, and reasonably foreseeable activities.

Table 6-4 of the Continued Storage GEIS summarizes the NRC staff's conclusions about the incremental impact of at-reactor storage, away-from-reactor storage, and the cumulative impacts of continued storage when added to other past, present, and reasonably foreseeable activities (NRC 2014-TN4117). The impact levels shown in Table 6-4 are denoted as SMALL, MODERATE, and LARGE as a measure of their expected adverse environmental impacts. Most impacts were found to be SMALL and SMALL to MODERATE. For some resource areas—such as terrestrial resources, environmental justice,[5] and climate change—the impact determination language is specific to the authorizing regulation, Executive Order, or guidance. Impact determinations that include a range of impacts reflect uncertainty related to both geographic variability and the temporal scale of the analysis. As a result, based on analyses performed in the Continued Storage GEIS, further site-specific analysis would be unlikely to result in impact conclusions with different ranges. The analyses of the Continued Storage GEIS were codified into 10 CFR 51.23 (79 FR 56238-TN4104).

Many of the assumptions provided in Section 1.8.3, Analysis Assumptions, of the Continued Storage GEIS and the Continued Storage GEIS's subsequent analysis are unaffected by the deployment and use of ATF, increased enrichment, and higher burnup levels. The principal analysis in the Continued Storage GEIS involves onsite impacts related to the siting, operating, and maintenance of an ISFSI and Dry Transfer System (DTS) facilities over all timeframes during continued storage (NRC 2014-TN4117). None of these assumptions would change due to the deployment and use of ATF because ISFSI and DTS facilities are sufficient to store ATF, including fuels with increased enrichment and higher burnup levels. For example, the waste management resource area involves radioactive and chemical wastes generated by the operation of the ISFSI and the DTS (e.g., used canisters, decontamination swabs, air filters, used personal protection equipment, and industrial practices involving the use of solvents or other chemicals) and does not directly involve the spent fuel in the storage casks. Only a select few topics considered in the Continued Storage GEIS have a connection with the spent fuel itself and how it could result in offsite environmental impacts, namely related to "Public and Occupational Health," "Postulated Accidents," and "Potential Acts of Terrorism." Even though the Continued Storage GEIS does discuss transportation of SNF, the transportation of spent ATF to a surrogate geologic repository is addressed in detail in Section 3 of this NUREG.

For public and occupational health, the NRC staff concluded in the Continued Storage GEIS that the radiological doses would be expected to continue to remain below the regulatory dose limits during continued storage and all of the related activities would have small environmental impacts (NRC 2014-TN4117). The NRC staff reached this conclusion in Sections 4.16 and 4.17 of the Continued Storage GEIS because the operations during continued storage would have a smaller workforce, lower volume of traffic and shipment activities, and continued storage represents a fraction of the activities occurring during reactor operations, as previously analyzed in the 2013 License Renewal GEIS (NRC 2013-TN2654) and in other NRC studies. This conclusion would not be different for spent ATF since the above discussion also applies to regulatory dose limits under similar operation-based conditions.

Regarding the analysis of postulated accidents in the Continued Storage GEIS (NRC 2014-TN4117), any spent ATF must be safely stored and decay heat must be appropriately removed once the spent ATF is removed from the reactor. This includes protection from and the mitigation of severe accidents, which are accidents that may challenge safety systems at a level higher than that for which they were designed. The concerns about severe accidents within an ISFSI, whether involving at-reactor or away-from-reactor storage, were analyzed in the Continued Storage GEIS (NRC 2014-TN4117). The lowest consequence events with any radiological release involved dropping a cask. The highest consequences were associated with an impact on the storage cask followed by a fire, such as could occur after an aircraft impact. In all cases, the NRC staff determined the likelihood of the event would be very low and the

[5] The NRC defines Environmental Justice as "The fair treatment of people of all races, cultures, incomes, and educational levels with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies." (NRC 2014-TN4117)

environmental risk of an accident would be small. The consequences described for cask drops at an ISFSI also provided some insight into the consequences of severe accidents in a DTS. Compliance with NRC regulations for spent fuel handling and storage would likely make the risk of severe accidents in a DTS small. In addition, the consequences of any severe accident in a DTS would likely be comparable to or less than that for the cask drop accident described above, mainly due to similarities in the inventory associated with casks and the waste form. This resulted in the NRC staff concluding in the Continued Storage GEIS that the likely impacts from activities in a DTS also would be small. Because the same NRC regulatory requirements for spent fuel handling and storage would apply, impacts from activities in an ISFSI or DTS with spent ATF would also be no different.

An assessment of the risks that could potentially result from acts of terrorism or radiological sabotage was also provided in the Continued Storage GEIS (NRC 2014-TN4117) and would still apply to spent ATF. The assessment was based, in part, on the analysis provided in the licensing of the Diablo Canyon Power Plant ISFSI and accounted for the security and protective measures required by NRC regulations (as described in Section 4.19 of the Continued Storage GEIS). The NRC staff determined that the potential for theft or diversion of LWR spent fuel from the ISFSI with the intent of using the contained special nuclear material for nuclear explosives is not considered credible because of the following:

- the inherent protection afforded by the massive, reinforced concrete storage module and the steel storage canister
- the unattractive form of the contained special nuclear material, which is not readily separable from the radioactive fission products
- the immediate hazard posed by the high radiation levels of the spent fuel to persons not provided with radiation protection

The NRC staff concluded in the Continued Storage GEIS (NRC 2014-TN4117) that for acts of terrorism, even though the environmental consequences of a successful attack could be large, the very low probability of a successful attack ensures that the environmental risk would be small for operational ISFSIs and DTSs during continued storage. Because the ISFSI infrastructure and the required physical protection would be no different for spent ATF than for existing SNF, the same considerations provided in the Continued Storage GEIS (NRC 2014-TN4117) of a very low probability of an accident or of a successful terrorist attack with the resulting small environmental risk would apply during continued storage of spent ATF. Finally, the Commission, in the Continued Storage rulemaking, reclassified the offsite radiological impacts of SNF and HLW disposal as a generic issue; no impact level was assigned and the entry under the column heading of Finding in Table B-1 in Appendix B of 10 CFR Part 51 was revised to address the existing radiation standards (79 FR 56238-TN4104).

Higher Burnup Appendix I of the Continued Storage GEIS provides background information about the licensing, storage, and transportation of high burnup uranium oxide fuel, such as in the case of ATF with increased enrichment and higher burnup (HBU) levels (NRC 2014-TN4117). As noted at the end of Appendix I of the Continued Storage GEIS, the environmental impacts do not require separate consideration of high burnup fuel because the unique characteristics of high burnup fuel are not a factor in environmental impact assessment for the resource areas considered in the Continued Storage GEIS.

As discussed in Section 2.1.1.3 of the Continued Storage GEIS, the use of high burnup fuel could create less spent fuel than a facility that uses low burnup fuel, while providing the same energy output. Therefore, for most resource areas evaluated in the Continued Storage GEIS, the impacts of storing high burnup fuel would be the same as or slightly less than the impacts associated with storing low burnup fuel. This is primarily because storing less spent fuel would require less land. This result is consistent with earlier published analyses of the environmental effects of high burnup fuel (Ramsdell et al. 2001-TN4545) that included the impacts from handling accidents, transportation, and onsite storage in support of environmental evaluations of operating NPPs.

Similarly, radionuclide inventories and thermal loading limits for ATF at higher burnup levels would not be a significant departure from the certified spent fuel shipping and storage containers. For example, the radionuclide inventory and related container shielding for any type of spent ATF must meet the regulatory requirements of 10 CFR 71.47 (TN301), “External radiation standards for all packages,” and 10 CFR 72.236 (TN4884), “Specific requirements for spent fuel storage cask approval and fabrication.” In addition, any shipping or storage containers for spent ATF would have to satisfy the regulatory requirements of 10 CFR 71.55 (TN301), “General requirements for fissile material packages,” and 10 CFR 72.236 (TN4884) “Specific requirements for spent fuel storage cask approval and fabrication,” which include the following:

- Confine fuel to a known volume.
- Ensure compliance with criticality safety.
- Meet specific structural testing requirements.
- Permit normal handling and retrieval.

Additionally, Section B.3 of the Continued Storage GEIS describes spent fuel degradation mechanisms that could occur during continued storage, which could also affect spent ATF. These include a mechanism (i.e., hydride reorientation) in which high burnup spent fuel cladding can become less ductile (more brittle) over time as cladding temperatures decrease. Taking actions (e.g., repackaging or providing supplemental structural support) can reduce risks posed by damaged fuel while maintaining fuel-specific or system-related safety functions. Further, as stated in Section B.3 of the Continued Storage GEIS, storage of spent fuel beyond the short-term storage timeframe would continue under an approved aging management program ensuring that monitoring and maintenance are adequately performed. This would also apply for high burnup spent ATF.

In conducting this generic analysis in the Continued Storage GEIS, the NRC staff applied conditions and parameter values that are sufficiently conservative to bound the impacts such that any variances that may occur from site to site are unlikely to result in environmental impact determinations that are greater than those presented in the Continued Storage GEIS. Therefore, since spent ATF would conform with the analysis of the Continued Storage GEIS (NRC 2014-TN4117), the Continued Storage GEIS would still be bounding for the environmental impacts of spent ATF.

2.3 Other Considerations

2.3.1 Consideration of Environmental Justice

As stated in NRC’s Policy Statement on the Treatment of Environmental Justice Matters in NRC Regulatory and Licensing Actions (69 FR 52040-TN1009),

> An NRC [environmental justice (EJ)] analysis would be limited to the impacts associated with the proposed action (*i.e.*, the communities in the vicinity of the proposed action). EJ-related issues differ from site to site and normally cannot be

resolved generically. Consequently, EJ, as well as other socioeconomic issues, are normally considered in site-specific EISs. Thus, due to the site-specific nature of an EJ analysis, EJ-related issues are usually not considered during the preparation of a generic or programmatic EIS. EJ assessments would be performed as necessary in the underlying licensing action for each particular facility.

The environmental impacts of various individual operating uranium fuel cycle facilities are addressed in separate EISs prepared by the NRC. These documents include analyses that address human health and environmental impacts to minority and low-income populations. Electronic copies of these EISs are available through the NRC's public Web site under Publications Prepared by NRC Staff document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC's Agencywide Documents Access and Management System (ADAMS) at https://www.nrc.gov/reading-rm/adams.html.

2.3.2 Greenhouse Gases

Table S-3 of 10 CFR 51.51(b) (TN250) does not provide an estimate of greenhouse gas (GHG) emissions associated with the uranium fuel cycle; it only addresses pollutants that were of concern when the table was promulgated in the 1980s. However, Table S-3 states that 323,000 MWh is the assumed annual electric energy use for the reference 1,000 MW(e) NPP and that this 323,000 MWh of annual electric energy is assumed to be generated by a 45 MW(e) coal-fired power plant burning 118,000 MT of coal. Table S-3 also assumes that approximately 135,000,000 standard cubic feet (scf) of natural gas is required per year to generate process heat for certain portions of the uranium fuel cycle. The NRC staff estimates that burning 118,000 MT of coal and 135,000,000 scf of natural gas per year results in approximately 253,000 MT of carbon dioxide equivalent (CO_2e) being emitted into the atmosphere per year because of the uranium fuel cycle (Harvey 2013-TN2646). This value of CO_2 emissions is with the assumption in WASH-1248 that all electricity use is provided by coal. Currently, coal produces 19.5 percent of all electricity, which corresponds to approximately 63,000 MWh, and natural gas produces 39.8 percent of electricity, which corresponds to about 128,600 MWh from burning approximately 946 million scf, while the remaining approximately 131,400 MWh is derived from non-CO_2 sources. Applying the analysis of Harvey (2013-TN2646) for the 323,000 MWh of electricity generation, coal generation would produce approximately 47,800 MT CO_2e, natural gas generation would produce approximately 51,660 MT CO_2e for a total from all sources (e.g., natural gas for process heat) for the uranium fuel cycle of approximately 107,200 MT CO_2e annual emissions. This CO_2e value is only about 42 percent of the Table S-3 CO_2e emissions. The U.S. Environmental Protection Agency (EPA) notes that in 2020, U.S. GHG emissions totaled 5,981 million MT CO_2e (EPA 2023-TN8681). Thus, the uranium fuel cycle contribution is a very small fraction of the U.S. GHG emissions.

As discussed above, the uranium fuel cycle generates substantially fewer GHGs today than it did when the agency issued WASH-1248 and Table S-3. Consequently, Table S-3 assumed that a coal-fired plant is used to generate the 63,000 MWh, and a natural gas-fired plant is used to generate 128,600 MWh of annual electric energy for the uranium fuel cycle. This power generation assumption results in conservative air emission estimates. Therefore, the NRC staff concludes that the values for electricity use and air emissions in Table S-3 continue to be appropriately bounding values. On this basis, the NRC staff concludes that the fossil-fuel impacts, including GHG emissions, from the direct and indirect consumption of electric energy for fuel cycle operations would be not significant.

2.4 Accident Tolerant Fuel Uranium Fuel Cycle Conclusions

Based on its review of the available information, the NRC staff concludes that the uranium fuel cycle involving ATF technologies with increased enrichment up to 10 wt% U-235 and higher burnup levels up to 80 GWd/MTU will have environmental impacts that are less than or comparable to those of current LWR fuels and less than those discussed in Table S-3. Lower front-end uranium fuel cycle environmental impacts than those provided in Table S-3 already exist for traditional fuel as the result of lower overall natural uranium extraction impacts (in-situ uranium recovery versus deep or pit mining and milling) and existing improvements in enrichment technologies (gaseous centrifuges versus gaseous diffusion enrichment). Improved reactor efficiencies (longer refueling times), and reduced waste and spent fuel inventories from the increased enrichment and higher burnup levels are also a factor in lowering the uranium fuel cycle environmental impacts than what has been considered for prior fuel cycle evaluations (e.g., as in the 1996 and 2013 versions of the License Renewal GEIS).

Regarding the deployment and use of ATF with increased enrichment and higher burnup levels, the NRC staff determined that the analyses in the Continued Storage GEIS were sufficiently conservative to bound the impacts such that any variances that may occur from site to site are unlikely to result in environmental impact determinations that are greater than those presented in the Continued Storage GEIS. Therefore, the NRC staff determined that spent ATF would conform to the analyses of the Continued Storage GEIS (NRC 2014-TN4117).

3 TRANSPORTATION

This section addresses the radiological and nonradiological environmental impacts from normal operating and accident conditions resulting from (1) shipment of unirradiated ATF to the NPP, (2) shipment of spent ATF to a postulated permanent geologic repository, and (3) shipment of LLRW and mixed waste generated through operations with ATF to a designated offsite disposal facility. For the purposes of these analyses, the NRC staff considered the proposed Yucca Mountain, Nevada, repository site as a surrogate destination for shipments to a permanent repository postulated in the western United States (to maximize estimated transportation impacts). This analysis would also apply for shipments to an interim storage facility with later shipments to the permanent geological repository.

3.1 Transportation Package Regulations

The NRC and the U.S. Department of Transportation (DOT) regulate the packaging and shipment of radioactive material by all transport modes in the United States. As presented in Section 1.4 of NUREG-2125 (NRC 2014-TN3231), DOT regulates the transportation of radioactive materials as part of hazardous materials transportation that are under 10 CFR 71.5 (TN301). Mode-specific regulations are described in 49 CFR Parts 174 to 177 and specifications for packaging are provided in 49 CFR Part 178 (TN5160). In addition, 49 CFR 173.471 (TN6622) allows the use of packages certified by the NRC under 10 CFR Part 71 (TN301), "Packaging and Transportation of Radioactive Material". The regulations of 10 CFR Part 20, "Standards for Protection Against Radiation" (TN283), also are relevant since they prescribe the largest allowable radiation dose that a member of the public may receive from NRC-licensed activities.

NRC transportation regulations apply to the approval and shipment of transportation packages. DOT regulations include labeling, occupational and vehicle standards, registration requirements, reporting requirements, and packaging regulations. Generally, DOT packaging regulations apply to industrial and Type A packaging, including excepted packages per 49 CFR 173.421, whereas the NRC regulations apply to fissile materials packages and Type B packages. Industrial and Type A non-fissile packages are designed to resist the stresses of routine transportation and are not designed to maintain their integrity in accidents, although many do. Type B packages are used to transport very hazardous quantities of radioactive materials, such as SNF. They are designed to maintain their integrity, prevent criticality, and provide radiation shielding in hypothetical accident conditions, because the NRC recognizes that any transport package and vehicle may be subject to the risks and impacts of traffic accidents.

U.S. transportation of radioactive material regulations are also consistent with those of the International Atomic Energy Agency (IAEA). The NRC has historically revised its transportation safety regulations of 10 CFR Part 71 (TN301) to ensure harmonization with the IAEA standards. Such changes in NRC regulations over time are necessary to maintain a consistent regulatory framework with DOT for the domestic packaging and transportation of radioactive material and to ensure general accord with IAEA standards.

3.2 NRC Regulations for Evaluating the Environmental Impacts from Transportation of Fuel and Waste

In accordance with 10 CFR 51.52 (TN250), a full description and a detailed analysis of transportation impacts is not required when licensing an LWR (i.e., impacts are assumed to be bounded by 10 CFR 51.52(c) [TN250], Summary Table S-4 – Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor [herein denoted as Table S-4]) if the reactor meets the following criteria:

- the reactor has a core thermal power level that does not exceed 3,800 MW(t)
- fuel is in the form of sintered uranium oxide pellets that have U-235 enrichment not exceeding 4 wt%, and the pellets are encapsulated in zircaloy-clad fuel rods
- the average level of irradiation of the fuel from the reactor does not exceed 33 GWd/MTU, and no irradiated fuel assembly is shipped until at least 90 days after it is discharged from the reactor
- with the exception of irradiated fuel, all radioactive waste shipped from the reactor is packaged and in solid form
- unirradiated fuel is shipped to the reactor by truck; while irradiated (spent) fuel is shipped from the reactor by truck, railcar, or barge; and radioactive waste, other than irradiated fuel, is shipped from the reactor by truck or railcar

The environmental impacts of the transportation of fuel and radioactive wastes to and from nuclear power facilities are resolved generically in 10 CFR 51.52 (TN250), provided that the specific conditions in the rule (see above) are met. The NRC may consider requests for licensed plants to operate at conditions above those in the facility's licensing basis; for example, higher burnups (above 33 GWd/MTU), enrichments (above 4 wt% U-235), or thermal power levels (above 3,800 MW[t]). Departures from the conditions itemized in 10 CFR 51.52(a) (TN250) are to be supported by a full description and detailed analysis of the environmental effects, as specified in 10 CFR 51.52(b) (TN250).

3.3 Table S-4 on the Transportation of Fuel and Waste

The NRC performed a generic analysis of the environmental effects of the transportation of fuel and waste to and from LWRs in WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants" (AEC 1972-TN22) and in a supplement to WASH-1238, NUREG–75/038 (NRC 1975-TN216) and found the impact to be small. These documents provided the basis for 10 CFR 51.52 (TN250) and the environmental impacts listed in Table S-4 of § 51.52(c). Table S-4 summarizes the environmental impacts of transportation of fuel and waste to and from one LWR of 3,000 to 5,000 MW(t) (1,000 to 1,500 MW[e]). The impacts of Table S-4 are for normal conditions of transport and accidents in transport for a reference 1,100 MW(e) LWR with 1-year refueling cycles. The environmental data in Table S-4 are applicable to LWRs that use uranium oxide, or UO_2, fuel that meets specific criteria in 10 CFR 51.52(a) (TN250), such as 4 wt% U-235 and irradiated fuel not to exceed 33 GWd/MTU. However, as discussed below, Addendum 1 of the 1996 License Renewal GEIS (NRC 1999-TN289) and Section 4.12.1.1, Uranium Fuel Cycle, of Revision 1 of the License Renewal GEIS (NRC 2013-TN2654), discuss extending Table S-4 conditions to bound LWR fuels with up to 5 wt% U-235 and burnup levels of up to 62 GWd/MTU.

As provided in Table S-4, dose to transportation workers during normal transportation operations was estimated to result in a collective dose of 4 person-rem per reference reactor-year. The combined dose to the public along the route and the dose to onlookers were estimated to result in a collective dose of 3 person-rem per reference reactor-year. Environmental risks of radiological effects during accident conditions, as stated in Table S-4, are small. Nonradiological impacts from postulated accidents were estimated as one fatal injury in 100 reference reactor-years and one nonfatal injury in 10 reference reactor-years.

Based on public comments on the 1996 version of the License Renewal GEIS (NRC 1996-TN288), the NRC reevaluated the transportation issues and the adequacy of Table S-4 for license renewal application reviews. In 1999, the NRC issued Addendum 1 of the License Renewal GEIS, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants Addendum to Main Report” (NRC 1999-TN289), in which the agency evaluated the applicability of Table S-4 to future license renewal proceedings, given that the spent fuel is likely to be shipped to a geologic repository (as opposed to several destinations, as originally assumed in the preparation of Table S-4) and given that shipments are likely to involve more highly enriched unirradiated fuel (more than 4 percent as assumed in Table S-4) and higher burnup spent fuel (higher than 33 GWd/MTU as assumed in Table S-4). In Addendum 1, the NRC staff published in 1999 the evaluation of the impacts of transporting the spent fuel from reactor sites to the then-candidate repository at Yucca Mountain and Ramsdell evaluated the impacts of shipping more highly enriched unirradiated fuel and higher burnup spent fuel (Ramsdell et al. 2001-TN4545). On the basis of the evaluations, the NRC concluded that the values provided in Table S-4 would still be bounding, as long as (1) the enrichment of the unirradiated fuel was 5 percent or less, (2) the burnup of the spent fuel was 62 GWd/MTU or less, and (3) the higher burnup spent fuel (higher than 33 GWd/MTU) was cooled for at least 5 years before being shipped offsite. A later study found that impacts presented in Table S-4, if not significantly affected by fission gas releases, do not change significantly with increasing burnup up to 75 GWd/MTU, provided that the fuel is cooled for at least 5 years before shipment (Ramsdell et al. 2001-TN4545).

3.4 Additional NRC Studies of Radioactive Material Transportation Risks

Since the publication of WASH-1238 (AEC 1972-TN22) and NUREG-75/038 (NRC 1975-TN216), the NRC has undertaken several studies regarding the risk from the transportation of radioactive material. Each study improved upon the assumptions and analysis techniques for assessing these risks compared to the prior studies.

In September 1977, the NRC published NUREG-0170, “Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,” which assessed the adequacy of the regulations in 10 CFR Part 71 (TN301), then entitled “Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions” (NRC 1977-TN417, NRC 1977-TN6497). In that assessment, the measure of safety was the risk associated with radiation doses to the public under routine and accident transport conditions, and the risk was found to be acceptable. Since that time, there have been two affirmations of this conclusion for SNF transportation, each using improved tools and information.

First, a 1987 study applied actual accident statistics to projected spent fuel transportation (Fischer et al. 1987-TN4105). This study, known as the “Modal Study,” recognized that accidents could be described in terms of the strains they produced in transportation packages (for impacts) and the increase in package temperature (for fires). Like NUREG-0170 (NRC 1977-TN417, NRC 1977-TN6497), the 1987 study based risk estimates on models because the limited number of accidents that had occurred involving spent fuel shipments was not sufficient to support projections or predictions. The Modal Study’s refinement of modeling techniques and use of accident frequency data resulted in smaller assessed risks than had been projected in NUREG-0170.

Second, as previously mentioned, in 1999 the NRC published Addendum 1 of the License Renewal GEIS (NRC 1999-TN289), which documents the NRC staff's analysis of the potential cumulative impacts of transporting SNF in the vicinity of a single high-level waste repository (then designated by the Nuclear Waste Policy Act of 1982 (H.R. 3809, Public Law 07-435) as being located at Yucca Mountain, Nevada) and summarizes the NRC staff's analyses undertaken to determine whether the environmental impacts of the transportation of higher enrichment and higher burnup SNF are consistent with the values of 10 CFR 51.52 (TN250), Table S-4. The intent of the study was to generically analyze the cumulative impacts associated with transportation of SNF as a result of NPP license renewal. On the basis of the evaluations, the NRC concluded that the values given in Table S-4 would still be bounding, as long as (1) the enrichment of the unirradiated fuel was 5 percent or less, (2) the burnup of the spent fuel was 62 GWd/MTU or less, and (3) the higher burnup spent fuel (higher than 33 GWd/MTU) was cooled for at least 5 years before being shipped offsite. Addendum 1 of the 1996 License Renewal GEIS was incorporated into the 2013 License Renewal GEIS.

In 2000, a study of two generic truck packages and two generic rail packages analyzed the package structures and response to accidents by using computer modeling techniques (Sprung et al. 2000-TN222). Even though more than 1,000 spent fuel shipments had been completed in the United States by the year 2000 and many thousands more had been completed safely internationally, there had been too few accidents involving spent fuel shipments to provide statistically valid accident rates. Therefore, the study used semi-trailer truck and rail accident statistics for general freight shipments. Sprung et al. 2000 (TN222) used improved technology to analyze the ability of containers to withstand an accident. This study concluded that the risk from the increased number of spent fuel shipments that could occur in the first half of this century would be even smaller than originally estimated in NUREG-0170 (NRC 1977-TN417, NRC 1977-TN6497).

As previously mentioned, a study conducted for the NRC by Pacific Northwest National Laboratory (PNNL) was published in 2001 in NUREG/CR-6703 about the environmental effects of extending fuel burnup above 60 GWd/MTU (Ramsdell et al. 2001-TN4545). The study indicates that there are no significant adverse environmental impacts associated with extending peak-rod fuel burnup to 62 GWd/MTU. Although the study evaluated the environmental impacts of fuel burnup up to 75 GWd/MTU, certain aspects of the review were limited to evaluating the impacts of extended burnup up to 62 GWd/MTU because of the need for additional data about the effect of extended burnup on gap-release fractions. For those aspects of the assessment in which the environmental impacts are not significantly affected by fission gas releases, the findings summarized by Ramsdell et al. (TN4545) indicate that there are no significant adverse environmental impacts associated with extending peak-rod fuel burnup to 75 GWd/MTU.

The most recent study, NUREG-2125, "Spent Fuel Transportation Risk Assessment," published in January 2014, presented the results of a fourth investigation into the safety of SNF transportation (NRC 2014-TN3231). The selected routes included origins and destinations analyzed in NUREG/CR-6672 (Sprung et al. 2000-TN222), thereby permitting the results of the studies to be compared. This investigation showed that the radiation emitted from the packages is a small fraction of naturally occurring background radiation and the risk from accidental release of radioactive material is less by several orders of magnitude than what was estimated in NUREG-0170. Because there have been only minor changes in the radioactive material transportation regulations described in NUREG-0170 (NRC 1977-TN417, NRC 1977-TN6497) and NUREG-2125 (NRC 2014-TN3231), the calculated dose from the external radiation from the package under routine transport conditions is similar to what was found in earlier studies. The improved analysis tools and techniques, improved data availability, and a reduction in uncertainty have made the estimate of accident risk from the release of radioactive material in NUREG-2125 approximately five orders of magnitude less than what was estimated in NUREG-0170. The analysis in NUREG-2125 estimated there is only about one-in-a-billion chance that an accident would result in a release of radioactive material. The results from NUREG-2125 (NRC 2014-TN3231) for spent ATF with increased enrichment and higher burnup levels are consistent with the environmental impacts associated with the transportation of fuel and radioactive wastes to and from current-generation reactors presented in Table S-4 of 10 CFR 51.52 (TN250).

Appropriate information from the above studies was applied regarding the deployment and use of ATF with increased enrichment and higher burnup levels in evaluating the environmental impacts from the transportation of fuel and wastes. Additionally, since WASH-1238 is the basis for Table S-4 and given that Ramsdell et al. (TN4545) was the last NRC study to assess environmental impacts from the transportation of fuel and waste with the maximum enrichment and burnup levels, this study evaluates the environmental impacts from the transportation of fuel and waste resulting from deployment and use of ATF in a manner that allows comparison of the study results to the prior assessments.

3.5 Transportation Impact Assessment Methodology

Radioactive material transportation risks are assessed for routine normal transportation conditions (incident-free) and accidents. For the assessment of impacts from normal conditions, risks are calculated for the collective populations of potentially exposed individuals. The accident assessment is where risks are calculated for the collective population living and working along the transportation route. This assessment includes the consideration of the probabilities and consequences of a range of possible transportation-related accidents, including low-probability accidents that have high consequences, and high-probability accidents that have low consequences.

The methodology for assessing transportation impacts is well developed and dates back to the 1970s with the analysis in NUREG-0170 applying the first version of the Radioactive Material Transport (RADTRAN) code (NRC 1977-TN417, NRC 1977-TN6497). RADTRAN, now NRC-RADTRAN, has been improved upon and extensively applied in several transportation studies (see above) and in numerous DOE and NRC environmental evaluations (e.g., various new nuclear facilities' environmental impact statements [EISs][1]). DOE's transportation risk assessment guidance is provided in DOE/EM/NTP/HB-01, "A Resource Handbook on DOE Transportation Risk Assessment," published in July 2002 (DOE 2002-TN418). NRC's guidance for a detailed transportation impact assessment is provided in Sections 3.8 and 5.7.2 of NUREG-1555 (NRC 2007-TN5141), and Section 7.4 of NUREG-1555 (NRC 1999-TN3548) for the NRC staff and Regulatory Guide 4.2, Revision 3, in Section 6.2 for NRC NPP licensees and applicants. The overall process is as follows:

- Set the transportation mode for each type of radioactive material. Unirradiated fuel is shipped to the reactor by truck; spent fuel is shipped from the reactor by truck, railcar, or barge; and radioactive waste other than irradiated fuel is shipped from the reactor by truck or railcar.
- Establish the transport package information for the material in question (unirradiated fuel, irradiated fuel, and radioactive waste) such as designating the certified package system with

[1] See NRC 2022-TN8072.

associated documentation concerning the packaging system capacity, approximate dimensions, radiation dose rates for the rated load, and weight. The packaging system's Certification of Compliance and Safety Analysis Report would provide this information.

- Determine the routes to be assessed based on the locations of fuel fabrication facilities and potential destinations for shipments of spent fuel and radioactive waste. Gather shipping route segment-specific values for a number of parameters (distances, population density, vehicle speed, traffic count, etc.) for the rural, suburban, and urban segments of the route. The code Web-Based Transportation Routing Analysis Geographic Information System (WebTRAGIS) can be a source for such information supplemented from other sources such as NRC-RADTRAN's technical manual and user guide, prior transportation analyses, and DOT databases.
- Collect the necessary information for assessing transportation accident risks. This includes a list of radionuclides with their package inventory values, severity probabilities, and release fractions, aerosolized fractions, and respirable fractions for the appropriate radionuclide chemical groups.

Section 3.6 and Appendices A, B, and D of this NUREG discuss in detail the data and information applied in this study with citations of their sources. Incident-free information was obtained from a variety of sources with the goal of locating and applying the most up-to-date values available from well-documented sources. Information related to accidents obtained from published NRC ATF studies by ORNL for radionuclide information at specified higher enrichment and burnup levels (see Appendix A) and Sprung et al. (TN222) was the principal source of transportation accident severity probabilities and release fractions.

3.5.1 Code Packages for Assessing Transportation of Fuel and Waste Risks

Radiological impacts of transportation of spent fuel were calculated by the NRC staff using the NRC-RADTRAN Version 1.0 computer code package with a graphical user interface (GUI). Routing and population data used in the NRC-RADTRAN calculations for truck shipments were obtained from the WebTRAGIS routing code (Peterson 2018-TN5839).

3.5.1.1 NRC-RADTRAN Version 1.0

NRC-RADTRAN Version 1.0 consists of RADTRAN Version 6.02.1 as the calculational driver code, based on the prior publicly available Version 6.02, in combination with a GUI to assist in data input and for performing calculations. RADTRAN Version 6.02.1 is a variation of RADTRAN Version 6.02 that has been modified for ease of use and for GUI compatibility. RADTRAN is a program for radioactive material transportation risk and consequence assessment that combines user inputs with physical and radiological data from its internal libraries and calculates radiological incident-free and accident risks and consequences. The detailed functionality of RADTRAN Version 6.02.1 is provided in the RADTRAN 6 Technical Manual (Weiner et al. 2014-TN3389) and instructions on the use of the GUI can be found in the NRC-RADTRAN Version 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073). RADTRAN was developed at Sandia National Laboratories (Sandia) and NRC-RADTRAN Version 1.0 with user guide and RADTRAN technical documentation is maintained by the NRC at the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) website (NRC 2022-TN8074).

NRC-RADTRAN can perform two separate and independent types of risk calculations. The incident-free analysis calculates the radiation dose from intact vehicles or packages, where the radiation dose is the dose from the radioactive materials within an intact transportation package as provided in the certificate of compliance (CoC). The accident analysis accounts for cask failure and dispersion of radionuclides, where the radiation dose is from the radionuclides released to the environment in the accident. Selecting incident-free analysis will disable the Accident, Radionuclide, Loss of Shielding, and Economic tabs, since they affect only the accident output from RADTRAN. Similarly, selecting accidental release analysis will disable the Stops and Handling tabs.

RADTRAN has changed over time, with the Version 5 (Neuhauser et al. 2000-TN6990; Neuhauser and Kanipe 2003-TN6989) being used in NRC environmental impact statements (EISs) published in the period 2006–2008, Version 5.6 (Weiner et al. 2008-TN302) being used in NRC EISs published in the period 2011–2016, and Version 6 being the current version (Weiner et al. 2013-TN3390, Weiner et al. 2014-TN3389). A specific example of how RADTRAN has changed over time is in how it estimates long-term doses after a transportation accident. RADTRAN Versions 5 and 5.6 estimated a long-term dose from transportation accidents based on 50 years of exposure to the radioactive material released from an accident, while RADTRAN Version 6 no longer provides these 50-year long-term dose estimates and instead provides dose estimates based on 1 year of exposure. Assuming that people are exposed for 50 years after an accident overestimates the doses from potential transportation accidents, and actual doses from transportation accidents would be much smaller due to effects of mitigation (e.g., relocation followed by cleanup of the radioactive materials).

3.5.1.2 WebTRAGIS

The routing code WebTRAGIS (Peterson 2018-TN5839) provides the necessary routing information that can be imported into NRC-RADTRAN, such as the one-way distance and the populations within 800 meters (m) (0.5 mile [mi]) for each side of a selected route. WebTRAGIS is deployed as a browser-based application interface, and the routing engine is located on a server at ORNL. WebTRAGIS offers users numerous options for route calculation using uniquely value-added network databases for highway, rail, and waterway infrastructures in the continental United States. The model also provides reporting information about population counts currently based on a combination of data sources, including 2010 U.S. Census Bureau block group population, American Community Survey intercensal, and other data sources for all transportation segments using the LandScan USA and LandScan Global population distribution data model adjusted to 2012 (Peterson 2018-TN5839).

WebTRAGIS determines routes from specified starting and ending points for highway, rail, or waterway transportation within the continental United States and provides the necessary information for each State traversed by a particular route. Routes are broken into “links,” or smaller segments of highway, railway, or waterway. WebTRAGIS derives route information around each network link along the transportation route, where link population densities and route distances are reported by rural, suburban, and urban categories. Various criteria for the route(s) to be determined may be specified, such as Highway Route Controlled Quantity criteria, which will be used for the SNF truck routes presented in this document. WebTRAGIS also has a setting for HAZMAT transportation because certain routes are unavailable to vehicles carrying HAZMAT. Nuclear fuel, regardless of whether it has been irradiated, is considered HAZMAT and therefore HAZMAT transportation settings would be enabled.

3.5.2 Normal Transportation Conditions

Normal conditions, sometimes referred to as “incident-free” transportation, are transportation activities during which shipments reach their destination without releasing any radioactive material to the environment (i.e., not being involved in a vehicular accident). Impacts from these shipments would be from the low levels of radiation that penetrate the shielding provided by shipping containers. Section 4.1.1 of the DOE handbook on transportation risk assessments discusses the typical methodology applied for normal, incident-free transportation risk assessments (DOE 2002-TN418).

Radiation exposures during normal conditions would occur to the following potentially exposed individuals: (1) persons residing along the ATF transportation route to or from the NPP site (i.e., the “off-link” population of residents); (2) persons at vehicle stops for refueling, rest, and vehicle inspections; (3) individuals in traffic traveling on the same route as an ATF shipment (i.e., “on-link” populations); and (4) transportation crew workers (i.e., drivers and package handlers). Figure 3-1 through Figure 3-4 demonstrate these radiation exposure scenarios. A description of the involved radiation exposure categories follows.

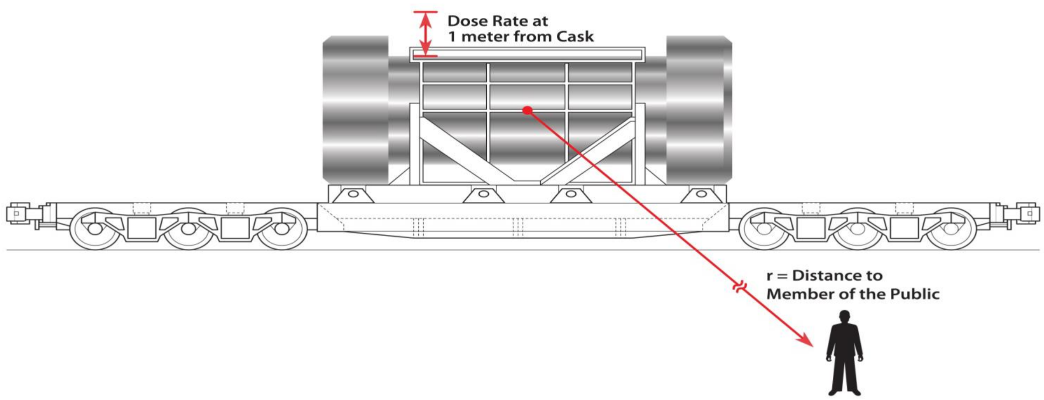

Figure 3-1 Diagrammatic Representation of Radiation-Based Exposure to Residents. (Source: Figures PS-1 and B-1 of NUREG-2125 [NRC 2014-TN3231])

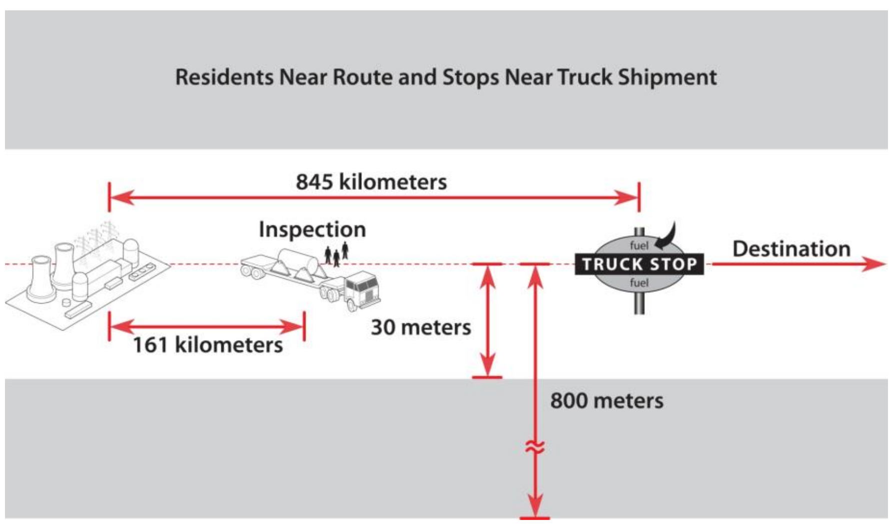

Figure 3-2 Diagram of a Truck Route as Modeled in NRC-RADTRAN (i.e., along the route). (Source: Figure B-2 of NUREG-2125 [DOE 2002-TN1236])

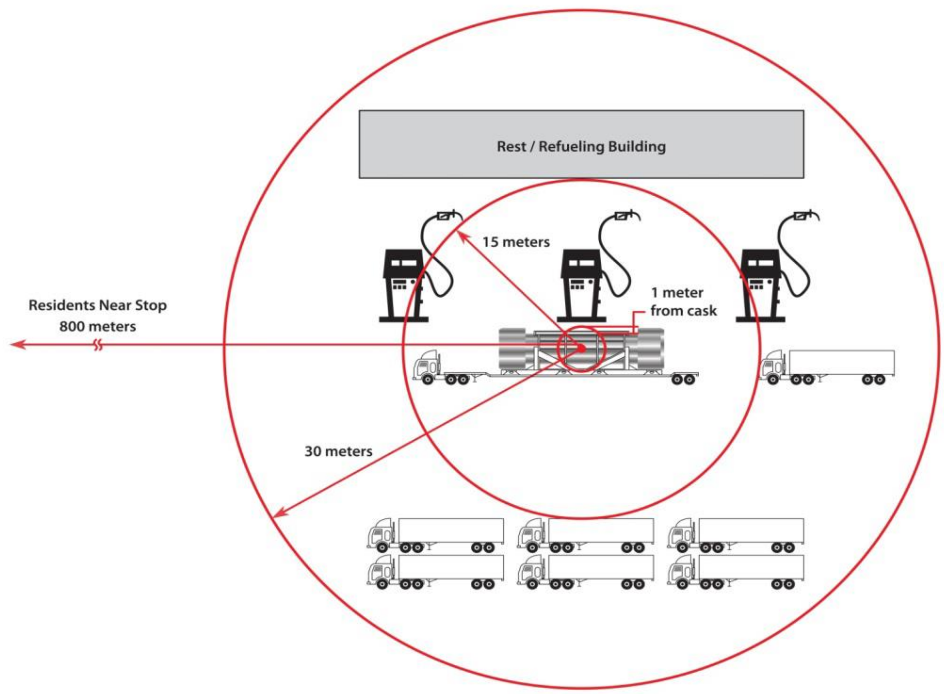

Figure 3-3 Diagram of Truck Stop Model (Not to Scale). (Source: Figures 2-10 and B-3 of NUREG-2125 [NRC 2014-TN3231])

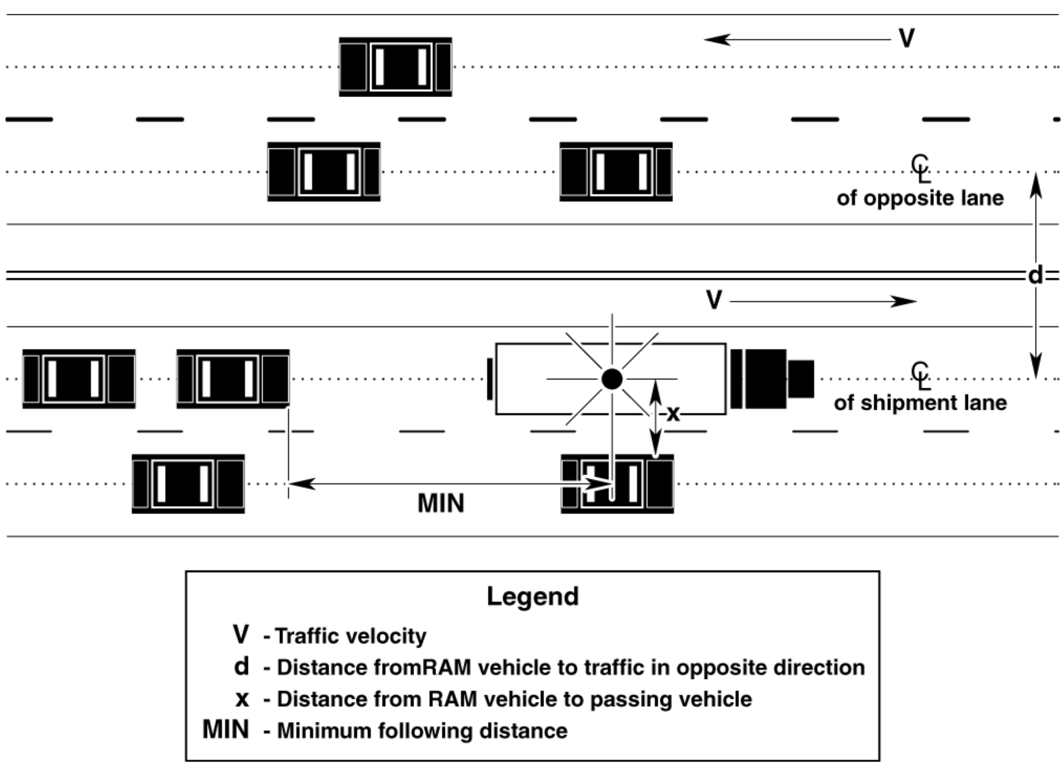

Figure 3-4 Illustration of Highway Traffic for Calculation of On-Link Dose (i.e., onlooker dose). (Source: Weiner et al. 2014-TN3389)

3.5.2.1 Persons Along the Route (Off-Link Population)

The analysis assumes that persons living or working on each side of a transportation route (i.e., within 800 m of the shipment route) would be exposed to all shipments along a particular route. The maximum exposed individual would occur under this category. The dose analysis for residents is based on data on the population density along the route and involves the application of U.S. Census data in the WebTRAGIS code, vehicle speeds, shielding, dose rates, and the number of times an individual may be exposed to a radioactive material shipment.

3.5.2.2 Persons at a Stop

All truck shipments to and from the NPP site are assumed to stop for refueling and food. They generally stop to refuel when half of the fuel is exhausted, based on one 30-minute stop per 4-hour driving time from the WebTRAGIS computer code (Peterson 2018-TN5839). Most truck stops are located in rural or suburban areas. Mandatory rest and crew changes are combined with refueling stops whenever possible. This scenario estimates doses to an employee and other members of the public at a service station where the exposure time and distance have been based on the observations discussed by Griego et al. (Griego et al. 1996-TN69). As shown in Figure 3-3, two regions at a stop are considered. The inner zone is in relation to those nearby the truck in a refueling area and related activities. The outer zone is regarding other members of the public who are also accessing the truck stop but are away from the truck shipment itself.

3.5.2.3 Person Sharing the Route (Onlookers or On-Link Population)

This exposure category addresses potential traffic conditions that could lead to a person being exposed to a loaded shipment while sharing the transportation route. Namely, as shown in Figure 3-4, this population includes persons traveling in the same or opposite direction as the shipment as well as persons in vehicles passing the shipment. Thus, individuals receive doses based on relative motion between their vehicle and the truck, setting the individual’s exposure time and distance. The NRC staff’s analysis assumed this exposure scenario would occur only one time to any individual.

3.5.2.4 Crew, Handlers, and Inspectors

Occupational doses from routine, incident-free radioactive materials transportation include doses to truck and train crew, railyard workers, inspectors, and escorts. Additionally, NRC-RADTRAN will also assess radiological exposures to package handlers at route origins and destinations as well as at transfer points (Weiner et al. 2014-TN3389). For this analysis, the NRC staff assumes that all ATF shipments are direct from the origin to the destination site with no intermediate transfer points.

Truck crew members (two per shipment) would receive the highest radiation doses during incident-free transport because of their proximity to the loaded shipping container for an extended period. The NRC analysis assumes that crew member doses are administratively controlled to 2 rem/yr, which is the DOE administrative control level presented in DOE-STD-1098-99, DOE Standard, Radiological Control, Chapter 2, Article 211 (DOE 2005-TN1235). The recommended limits are a 5-year effective dose of 2 rem/yr with no more than 5 rem in a single year (Friedberg and Copeland 2003-TN419). This limit is anticipated to apply to SNF shipments to a disposal facility because DOE would take title to the spent fuel at the reactor site using radiologically trained Federal or contracted drivers and would be responsible for delivering the

SNF shipments. While shipments to a licensed consolidated interim storage facility (CISF) could be performed by a non-DOE shipper, the 2 rem/yr dose to a crew member is still a reasonable assumption. As a result of this recommendation, a 2 rem/yr dose to truck crews is a reasonable estimate to apply to shipments of ATF.

Handlers are workers who guide the crane to the proper orientation for transportation packages both to pick up the cask and to lower it into position on the vehicle. The handlers also include a spotter and workers who lock and check the tiedowns after the package is in place. There may be more than five individuals involved but no more than five handlers are in proximity to the package at any given time. The standardization of handling equipment means there is little variation in this value in normal operations. Radioactive shipments are inspected by Federal or State vehicle inspectors, for example, at State ports of entry. Thus, inspectors would be near the package and exposed to the external radiation field around a package in the same manner as handlers.

3.5.2.5 NRC-RADTRAN Modeling of Normal Conditions

The modeling of radiation exposures within the NRC-RADTRAN code package for normal, incident-free conditions is presented in Section 2 of the RADTRAN 6 Technical Manual (Weiner et al. 2014-TN3389). This document has been incorporated in this study by reference.

3.5.3 Accident Conditions

Accident risks are a combination of accident frequency and consequence. When addressing accident risks from the transportation of fuel and waste, two components must be considered: radiological risks and nonradiological risks.

As discussed in Section 3.4, the NRC has conducted several transportation risks studies, generally concerning the radiological risks from SNF shipments. The process for assessing transportation risks is well established and documented, such as that described by Sprung et al. (2000-TN222) and in the DOE handbook on transportation risks (DOE 2002-TN418). Both documents provide a methodology road map for assessing transportation accidents and provide further details or guidance necessary to complete such an assessment with the RADTRAN code. Event trees are included for various potential transportation accidents, with severity levels and associated radioactive material release fractions for PWR and BWR spent fuel for truck and rail transportation packages. Much of the accident scenario information provided by Sprung et al. (2000-TN222) for accidents that exceed the regulatory hypothetical accident conditions of 10 CFR 71.73 (TN301) has been applied in this study and, therefore, it is incorporated by reference.

Nonradiological risks are the physical, nonradiological human health impacts projected to result from traffic accidents involving shipments of fuel and waste that do not consider the radiological or hazardous characteristics of the cargo. These risks can be viewed as “vehicle-related” risks due to being from mechanical causes. Nonradiological risks are based on the projected number of traffic accidents, injuries, and fatalities that could result from shipments to the NPP and return shipments of empty containers from the NPP. These nonradiological risks are calculated by multiplying the total distance traveled in each State by the appropriate State rate for transportation-related fatalities and injuries.

Nonradiological impacts are calculated using accident, injury, and fatality rates from published sources. The rates (i.e., impacts per vehicle-km traveled) are then multiplied with the estimated travel distances for workers and materials.

3.5.4 Data and Information Needs

Several guidance documents outline and discuss the necessary data and information for performing transportation of fuel and waste evaluations using the NRC-RADTRAN computer code. These guidance documents include the following:

- DOE/EM.NTP/HB-01, "A Resource Handbook on DOE Transportation Risk Assessment" (DOE 2002-TN418);
- NUREG-1555, "Environmental Standard Review Plan: Standard Review Plans for Environmental Reviews for Nuclear Power Plants," Sections 3.8 (NRC 2007-TN5141) and 7.4 (NRC 1999-TN8080);
- Regulatory Guide 4.2, Revision 3, "Preparation of Environmental Reports for Nuclear Power Stations," Section 6.2 (NRC 2018-TN6006);
- SAND2013-0780, "RADTRAN 6 Technical Manual" (Weiner et al. 2014-TN3389);
- SAND2013-8095, "RADTRAN 6/RadCat 6 User Guide" (Weiner et al. 2013-TN3390); and
- ERI/NRC 20-208, "NRC-RADTRAN 1.0 Quick Start User's Guide" (Ball and Zavisca 2020-TN8073).

The NRC-RADTRAN GUI input file editor, as described by Ball and Zavisca (TN8073), breaks down the data and information requirements by tabs (Ball and Zavisca 2020-TN8073), namely the following:

- vehicles
- links
- stops
- handling
- packages
- accidents
- radionuclides
- loss of shielding
- economic model
- default parameters

The NRC-RADTRAN calculations and necessary data inputs will depend on the desired analysis. Vehicle input data are required for calculations of both incident-free and accident doses. Package input data are optional for incident-free calculations and are required for calculating accident consequences. Several of the vehicle and package input data requirements can be obtained from the selected transport package's CoC and its Safety Analysis Report (e.g., dimensions, gamma and neutron fractions, and dose rates 1 m from the package surface).

Link input data can be obtained from WebTRAGIS for route-specific inputs (e.g., length, population density by rural, suburban, and urban zones by State). The population data currently applied by the WebTRAGIS code is based on the 2010 U.S. Census, the American Community Survey intercensal, and other sources (Peterson 2018). This results in a population density adjusted to 2012 as shown in the WebTRAGIS output files (Peterson 2018-TN5839). For this study, the code population density data are further adjusted using a population correction factor to account for the year 2022 population based on the 2020 U.S. Census and other sources. Traffic density data by State can be obtained from the RADTRAN 6/RadCat 6 User Guide tables in Appendix D (Weiner et al. 2013-TN3390) if State databases are not readily available. The Link data tab is also the location at which to enter the accidents per distance, which could be derived from published traffic accident, injury, and fatality data from DOT databases. These databases include the Federal Motor Carrier Safety Administration (FMCSA) for truck shipments (FMCSA 2022-TN8075) or past transportation studies (e.g., Saricks and Tompkins [1999-TN81] as adjusted by Blower and Matteson [2003-TN410] for truck shipments, and Abkowitz and Bickford (TDEC 2017-TN5261) for rail shipments).

Data inputs for the Stop and Handling tabs, such as distances and time are best obtained from published studies such as the Sandia study by Griego et al. (1996-TN69), NUREG/CR-6672 (Sprung et al. 2000-TN222), DOE/EIS-0250—namely the Yucca Mountain FEIS (DOE 2002-TN1236), and the WebTRAGIS User's Manual (Peterson 2018-TN5839).

The most common information source for the Accident tab is from NUREG/CR-6672 (Sprung et al. 2000-TN222) as supplemented by later studies (e.g., Mills et al. 2006) for the conditional probability by severity level and release fractions by chemical groups (i.e., particulates, gases, ruthenium, cesium, and crud). For the related Radionuclide tab, its data source is derived from a radionuclide inventory calculation for a specific type of nuclear fuel based on several factors like the power history for the NPP from a computer code such as ORIGIN or SCALE (Rearden and Jessee 2018-TN8282). Appendix A discusses the development of the radionuclide inventory applied in this study using computer codes associated with the SCALE code package. Another NRC-RADTRAN tab related to a specific type of vehicle accident is the Loss of Shielding tab with past studies being the best sources for data or other information about this type of accident event, such as Sprung et al. (2000-TN222), NUREG-2125 (2000-TN222), NUREG-2125 (NRC 2014-TN3231), and Weiner et al. (2014-TN3389).

For the optional Economic Model tab, the default values are listed in the RADTRAN 6/RadCat 6 User Guide (Weiner et al. 2013-TN3390). Additional economic modeling details are described in SAND2007-7120 (Osborn et al. 2007-TN8078).

The Default Parameters tab includes a large number of inputs, all of which are optional. Besides the help menu within the NRC-RADTRAN GUI, an analyst can find more detailed descriptions for several of the default parameters in the RADTRAN 6/RadCat 6 User Guide (Weiner et al. 2013-TN3390).

Data sources for nonradiological risks for State accident, injury, and fatality rates would be from publicly available Federal or State databases, such as FMCSA-published information through the Motor Carrier Management Information System (FMCSA 2022-TN8075).

Given the extent of the data and information necessary to properly perform a transportation risk assessment with NRC-RADTRAN, it must be emphasized that it is the responsibility of each analyst to ensure the appropriateness of all data and information being applied in the analysis. For further information about the input parameter values used in the NRC-RADTRAN calculations for a single shipment, see Appendix D.

3.6 Transportation Scenario Development

This section discusses the development of the ATF (with increased enrichment and higher burnup levels) transportation scenarios and related assumptions to be analyzed with the NRC-RADTRAN code. First, past NRC studies have analyzed transportation of fuel and waste impacts from both truck and rail shipments. This study aims to do the same. However, the previous analyses have demonstrated that truck shipments have larger impacts than rail shipments principally due to the larger number of truck shipments than rail due to the lower truck load capacities. Therefore, the principal analysis of this study will focus on truck shipments with rail shipments as a sensitivity case. Second, this study aims to assess the appropriateness of Table S-4 regarding the deployment and use of ATF with increased enrichment and higher burnup levels along with comparisons of impacts to those identified in past studies such as Ramsdell et al. (2001-TN4545). To support such a comparison, the assumptions and characteristics were selected to allow for the best direct comparison to WASH-1238 (AEC 1972-TN22) results as practicable. Therefore, this section discusses the selection of shipment origination and destination sites with corresponding shipping routes, transport package characteristics, and radionuclide inventory based on a maximum enrichment and burnup level.

3.6.1 Site and Route Selection

The characteristics of specific shipping routes (e.g., population densities, shipping distances) influence the normal radiological exposures. To address the differences that arise from the specific reactor site from which the spent fuel shipment originates, NPP sites were selected based on the four NRC regions. Representative reactor sites in each region were selected to illustrate the impacts of transporting spent ATF from a variety of possible locations. The NRC regions and the representative reactors selected for each region are as follows:

- Region I – Millstone Power Station (PWR)
- Region II – Turkey Point Nuclear Generating Units (PWR), Brunswick Steam Electric Plant (BWR)
- Region III – Enrico Fermi Nuclear Generating Station Unit 2 (BWR) and Dresden Nuclear Power Station (BWR)
- Region IV – Columbia Generating Station (BWR)

Out of these six sites, four are the same sites analyzed by Ramsdell et al. 2001 (TN4545), namely Brunswick Steam Electric Plant, Millstone Power Station (Millstone), Turkey Point Nuclear Generating Units (Turkey Point), and the WNP-2 site, which is now known as the Columbia Generating Station (Columbia). Enrico Fermi Nuclear Generating Station (Fermi) Unit 2 and Dresden Nuclear Power Station (Dresden) replace the now closed Zion NPP site used by Ramsdell et al. (TN4545). To allow for potential comparison of this study's results with the results of Ramsdell et al. (TN4545) these particular sites were selected. For each site, both BWR and PWR spent ATF shipments are considered and evaluated for the purpose of impact comparison owing to the different release fractions for BWR and PWR fuel designs, as shown in Table 7.31 of Sprung et al. (2000-TN222).

This study evaluates potential shipments of spent ATF to a postulated geologic repository in the western United States. For the purposes of this evaluation, the NRC staff considered the proposed Yucca Mountain, Nevada, geologic repository site (Yucca Mountain) as a surrogate destination for a permanent repository.[2] While the history of the proposed Yucca Mountain site and actions under the Nuclear Waste Policy Act of 1982 are well known, this site was used as a surrogate destination for spent ATF shipments because routes from U.S. East Coast sites would likely yield the highest impacts due to the involved distance and population centers the routes would travel through or be nearby. Their shipment distances would also be greater than spent ATF shipments to either of the currently licensed CISFs, for which the Interim Storage Partners site near Andrews, Texas, and the Holtec International site in Lea County, New Mexico, have been issued an NRC license (NRC 2021-TN7986, NRC 2023-TN8284). Additionally, the proposed Yucca Mountain site is the same destination site in some of the other NRC transportation studies and NRC new reactor EISs. The spent ATF routes must meet the DOT regulations for shipments of Highway Route Controlled Quantity of radioactive material, where such a highway route designation is an option within WebTRAGIS. The resulting spent ATF highway routes for each NPP site to the vicinity of the Yucca Mountain site are shown in Figure 3-5 for truck shipments and Figure 3-6 for rail shipments. Route distances are provided in Table C-2 of Appendix C.

For unirradiated ATF shipments, given that the radiological component is very low from the enriched uranium, a single route is considered representative of the potential nonradiological impacts. The originating fuel fabrication facility site to a NPP with the greatest shipping distance was selected for this part of the evaluation. This route would be from Framatome FFF near Richland, Washington, to the Turkey Point site of approximately 3,187 mi, or 5,129 km, as shown in Figure 3-5.

3.6.2 Package and Shipping Characteristics

Robust shipping packages are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR Part 71 (TN301). Spent fuel shipping packages must be certified Type B packaging systems, meaning they must withstand a series of postulated accident conditions with essentially no loss of containment or shielding capability in accordance with 10 CFR Part 71, Subpart E, and specifically after being subject to the tests in 10 CFR 71.73 (TN301). These packages also are designed with fissile material controls to ensure that the spent fuel remains subcritical under normal and accident conditions. As discussed in Section 1.5 and shown in Tables 1-1, A-1.1, and A-1.2 of NUREG-2125 (NRC 2014-TN3231), a number of Type B transport packages can be used for shipments of spent fuel, including spent ATF. Most of these Type B packages, especially for the rail packages, include an inner sealed SNF canister. This involves placing the spent fuel assembly into a canister while in the spent fuel pool, removing water from the canister, welding it closed, and then placing the canister into the Type B package. One result of this kind of packaging, specifically discussed in NUREG-2125 (NRC 2014-TN3231), is that radioactive material would not be released in an accident since it would remain contained in an inner welded canister inside the transport package. Only rail transport packages without inner welded canisters would release radioactive material and only then in exceptionally severe accidents (NRC 2014-TN3231). As is discussed later, this has an impact on the selection of the transport package to apply in the transport calculations.

[2] There is the potential for spent ATF to be shipped to an interim storage facility and, at a later time, to a geologic repository. Due to the location of shipment origins and the assumed surrogate geologic repository applied in this study, shipments to an interim storage facility and later to a geologic repository would not be appreciably different for the route considered in this study.

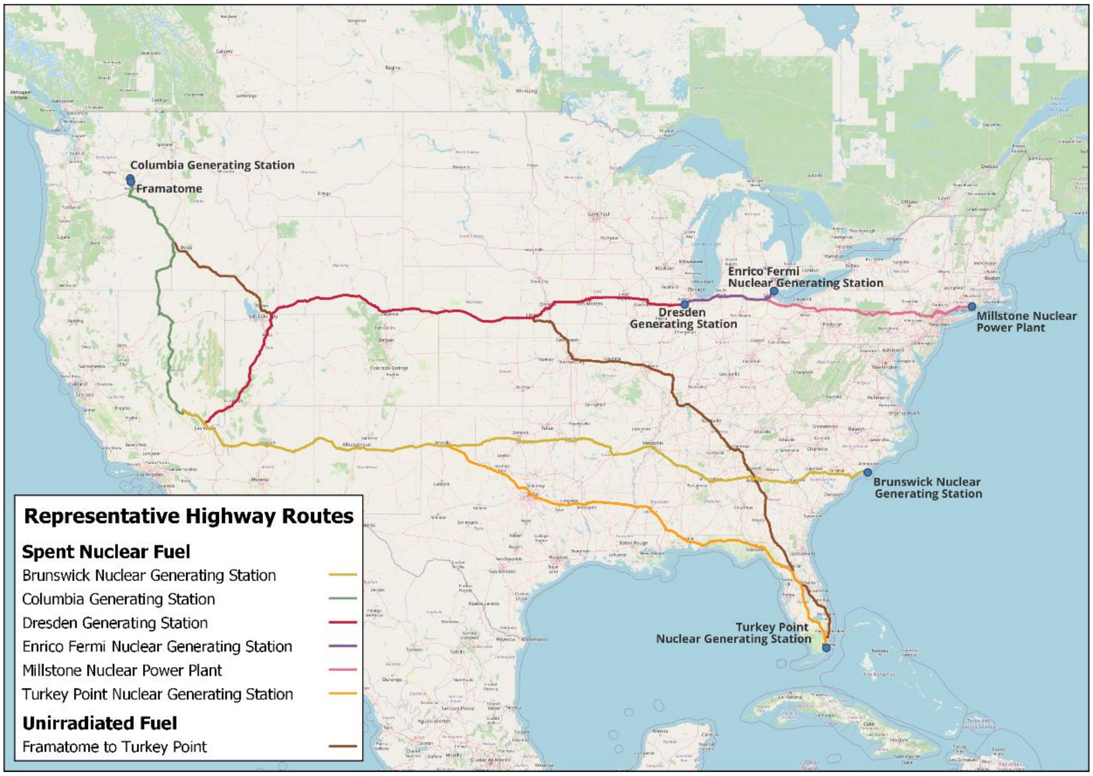

Figure 3-5 Highway Routes Across the United States

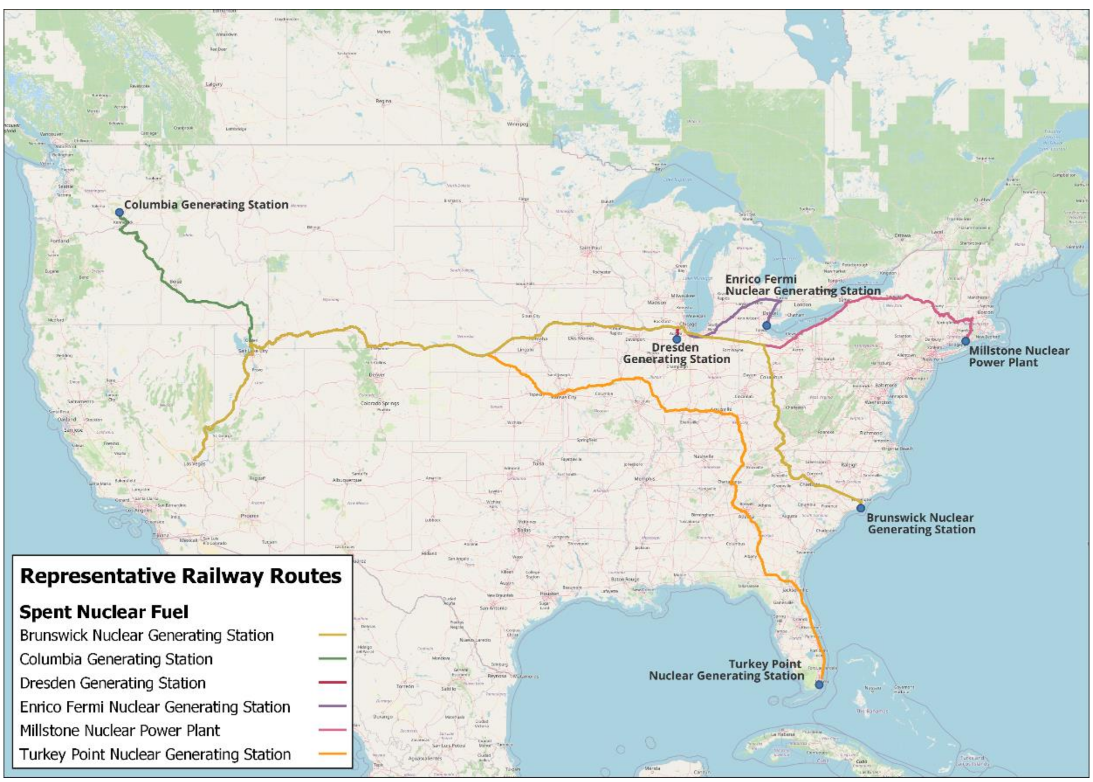

Figure 3-6 Rail Routes Across the United States

Because this study performs an evaluation to compare spent ATF transportation impacts to the impacts provided in 10 CFR 51.52(c) (TN250), Table S-4, the Type B package selected for this study would be as close to the kind of spent fuel package applied in WASH-1238 (AEC 1972-TN22) to allow for as direct of a comparison as is practicable. At the time of the WASH-1238 study (AEC 1972-TN22), there was only one approved design for a package that had sufficient length, cavity diameter, shielding, and heat dissipating capacity to be used for transporting irradiated fuel assemblies from nuclear power reactors. Namely, a truck package that could carry from one to three PWR spent fuel assemblies or from two to seven BWR spent fuel assemblies. Another factor for selecting a specific spent fuel package is that the selection must be consistent with the type of package, source term severity fractions, and release fractions being applied in the transportation calculation, namely those mentioned in Table 7.31 of Sprung et al. (2000-TN222). Given these considerations, this study selected the NAC International-Legal Weight Truck (NAC-LWT) package for truck spent ATF shipments and the NAC-Storage Transport Cask (NAC-STC) for rail spent ATF shipments. Important technical specifications for the NRC-RADTRAN calculations for each of these packages are provided in Table 3-1.

Table 3-1 NAC International-Legal Weight Truck (NAC-LWT) and NAC-Storage Transport Cask (NAC-STC) Technical Specifications

Technical Specification	NAC-LWT[(a)]	NAC-STC[(b)]
Fuel Assembly Capacity	PWR – 1 BWR – 2	PWR only – 26
Maximum Decay Heat per Assembly (kW)	PWR – 2.5 BWR – 1.1	0.85
External Dose Rate Toward Crew (mrem/hr)	0.72	2.70
External Dose Rate Toward Handlers and the Public (mrem/hr)	8.14	9.50

BWR = boiling water reactor; kW = kilowatt(s); NAC-LWT = NAC International-Legal Weight Truck; NAC-STC = NAC International-Storage Transport Cask; PWR = pressurized water reactor.
(a) NAC-LWT Safety Analysis Report, Revision 44, Volume 2 or 3, Part 4 of 5 (NI 2015-TN8076).
(b) NAC-STC Safety Analysis Report, Revision 18, Part 2 or 2 (NI 2017-TN8077).

Unirradiated ATF shipments would generally be made using commercial trucks that carry in the range of 10 to 14 unirradiated fuel transportation packages.

An example of this type of package for PWR fuel is the Traveller package (CoC 9380) that holds one PWR fuel assembly and the RAJ-II package (CoC 9309) with a capacity for 2 BWR fuel assemblies. Table S-4 includes a condition that the truck shipments would not exceed 73,000 lb as governed by Federal or State gross vehicle weight restrictions; the current DOT gross vehicle weight limit is 80,000 lb (23 CFR Part 658-TN8088). Based on these factors, this evaluation will set the number of unirradiated ATF assemblies in a truck shipment to 10 PWR assemblies in Traveller packages as shown in Figure 3-7 and 24 BWR assemblies in 12 RAJ-II packages as shown in Figure 3-8.

Figure 3-7 Unirradiated Pressurized Water Reactor Fuel Shipment Using Traveller Packages. (Source: Photo provided by the Westinghouse Electric Company, LLC [NRC 2022-TN8089])

Figure 3-8 Unirradiated Boiling Water Reactor Fuel Shipment Using RAJ-II Packages. (Photos courtesy of Global Nuclear Fuels, General Electric)

3.6.3 Number of Annual Unirradiated and Spent Accident Tolerant Fuel Shipments

The number of annual unirradiated and spent ATF shipments is dependent on the number of fuel assemblies that are required to complete a refueling outage. The NPPs in the United States typically shut down to refuel every 18 to 24 months. During a refueling outage, about one-third of the oldest fuel assemblies are removed from the core and placed in the spent fuel pool. The remaining two-thirds of fuel assemblies are reshuffled, and a batch of new, unirradiated fuel assemblies is added to the core to complete the refueling operation. This type of operation has also been called a “batch reload.” Additionally, industry has indicated that consideration is being given to batch reloads of half of the core (NEI 2023-TN9602). Herein the nomenclature of “one-third-core reloads” and “half-core reloads” will be applied to designate these reloading operations.

The numbers of shipments of fuel and waste were estimated in WASH-1238 on the basis of the shipments anticipated from a typical 1,100 MWe PWR. Table 1 of WASH-1238 has estimates of 6 unirradiated fuel shipments and 60 spent fuel shipments by truck for one-year core reloads (AEC 1972-TN22). With a 2-year refueling cycle, this would result in three unirradiated fuel shipments and 30 spent fuel shipments per year for one-third-core reloads. The corresponding number of shipments for half-core reloads every 2 years would be approximately 5 unirradiated fuel shipments per year and 45 spent fuel shipments per year. For an existing PWR, such as the AP1000 PWR at a nominal net electrical output of 1,110 MWe (same as in WASH-1238), the core loading is 157 fuel assemblies (Westinghouse 2011-TN261). This would result in a one-third-core reload of approximately 53 fuel assemblies and a half-core reload of approximately 78 fuel assemblies. Thus, an AP1000 with a 2-year refueling cycle, having 10 unirradiated fuel assemblies per shipment with the Traveller package, would require approximately 3 unirradiated fuel shipments per year for one-third-core reloads and approximately 4 unirradiated fuel shipments per year for half-core reloads. With one spent fuel assembly per shipment with the NAC-LWT package, an AP1000 would require approximately 27 spent ATF shipments per year for one-third-core reloads and approximately 39 spent fuel shipments per year for half-core reloads. For both unirradiated fuel and spent fuel, the number of shipments analyzed in WASH-1238 would bound the necessary shipments for an AP1000, a representative PWR. Therefore, the number of shipments from WASH-1238 will be applied in this evaluation as reasonable values for PWRs.

For a BWR rated at the same 1,100 MWe as in WASH-1238, the core loading is 624 fuel assemblies (Constellation 2022-TN8102), resulting in a one-third-core reload of approximately

208 fuel assemblies and a half-core reload of approximately of 312 fuel assemblies. Based on a 2-year refueling cycle and use of the RAJ-II and NAC-LWT packages (each holds two BWR fuel assemblies for truck shipments), there would be approximately four unirradiated fuel shipments and 52 spent fuel shipments per year for one-third-core reloads and 6 unirradiated fuel shipments and 78 spent fuel shipments per year for half-core reloads. Therefore, since the size of the core reloads for a BWR in WASH-1238 is 1/5th of the core, the above numbers for a larger core reload would provide for larger impacts and will be applied in this evaluation as reasonable values for BWRs.

The outcome of the deployment and use of ATF with increased enrichment and higher burnup levels will increase the time between refueling to be consistently once every 2 years for all NPPs. Since the impacts provided in Table S-4 are on a per reactor-year basis, this study produced results on a per reactor-year basis. The numbers of annual unirradiated and spent fuel shipments to be applied in this study are shown in Table 3-2. Additional details for the determination of the values in Table 3-2 are provided in Appendix D, Section D.3. This analysis also includes the return of the packages to the originating site to fully account for nonradiological impacts.

Table 3-2 Boiling Water Reactor and Pressurized Water Reactor Annual Unirradiated Fuel and Spent Fuel Shipments by Truck

Fuel Type	Boiling Water Reactor with One-Third-Core Reload	Pressurized Water Reactor with One-Third-Core Reload	Boiling Water Reactor with Half-Core Reload	Pressurized Water Reactor with Half-Core Reload
Unirradiated fuel	4	3	6	5
Spent fuel	52	30	78	45

Another key assumption for this analysis is that all spent ATF would move by legal weight truck rather than by rail or by a combination of rail and truck to reach the Yucca Mountain surrogate geologic repository. This is consistent with the conservative assumptions made in the evaluation of the environmental impacts of transportation of spent fuel presented in Addendum I to the License Renewal GEIS (NRC 1999-TN289). However, there are certified rail packages for shipping spent fuel and rail transport is an alternative to truck transport. As discussed in Addendum 1, these assumptions are conservative because the alternative assumptions involve rail transportation or heavy-haul trucks, which would reduce the number of spent fuel shipments. To verify and demonstrate this condition still holds, a sensitivity calculation based on rail shipment of PWR spent fuel is included in this study. This sensitivity calculation is based on the previously cited NAC-STC package that can hold 26 spent PWR fuel assemblies. Thus, there would be approximately 1.25 annual spent fuel shipments by rail given a one-third-core reload of 60 fuel assemblies and approximately 1.73 annual spent fuel shipments given a half-core reload of 90 fuel assemblies with a 2-year refueling frequency.

3.6.4 Fuel Characteristics and Radionuclide Inventory Based on Enrichment and Burnup

For near-term deployment and use of ATF, the nuclear industry is likely to pursue coated cladding or doped uranium oxide pellets. Even though it is not tied to accident tolerance, key aspects of deploying ATF involve the use of ATF uranium oxide pellets that have increased enrichment and have capability to reach higher burnup levels. By enhancing such fuel characteristics, licensees could extend the refueling cycle time to at least 2 years, which is longer than previously assessed with respect to WASH-1238 and Table S-4. Another consideration for this evaluation is the spent fuel carried by legal weight trucks consists of a single package with 0.5 MTU of spent fuel. This MTU value was applied in WASH-1238 as shown in that study's Appendix B, Table 1 (AEC 1972-TN22).

To also be aligned with prior transportation of spent fuel assessments, this evaluation assumes that ATF with increased enrichment and higher burnup levels can be transferred out of a spent fuel pool after 5 years of cooling for dry storage or placed into a certified transportation package. A key parameter for this is the heat load in a spent ATF assembly. While the actual time spent ATF would need to be kept under water in a spent fuel pool would be determined at the time of its removal from the reactor, its storage will depend on whether the conditions of the spent ATF assembly meet all conditions for dry (i.e., air cooling) storage or shipment. The principal impact of the 5-year cooling time after removal from a reactor is to set the radionuclide inventory within a spent ATF assembly as determined by an appropriately validated depletion computer code (e.g., SCALE, see Appendix A).

As discussed in Appendix A, the NRC staff relied on two studies performed by ORNL for use in this study of the radionuclide curie content, heat load, and MTU that were based on various enrichments, burnup levels, and a cooling time of 5 years. The ORNL studies did not assess enrichment levels above 8 wt% U-235 and above assembly averaged burnups of 80 GWd/MTU. Thus, these enrichment and burnup levels are the maximum values that could be addressed at this time for Table S-4 applicability. As discussed in Section 1.4, the use of near-term ATF technology is not expected to affect the fuel pellet source term, and any additional source term in the FeCrAl cladding itself is not dispersible. The resulting bounding fuel characteristics and composite radionuclide inventory parameter values applied in this study are shown in Table 3-3 and Table 3-4 (Note: Within this document, scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02).

Table 3-3 Fuel Parameter Values

Fuel Parameter	Value
Maximum Enrichment (weight percent uranium-235)	8
Maximum Burnup Level (GWd/MTU)	80
Assembly Heat Load (kW per package)	1 PWR fuel assembly — 2.39 2 BWR fuel assemblies — 1.03
MTU per package	0.5

GWd/MTU = gigawatt day(s) per metric ton of uranium; kW = kilowatt(s).

Table 3-4 Radionuclide Inventory Parameter Values

Element	Radionuclide Inventory (Curies)[(a)]
Co-60	4.38E+03
Kr-85	8.04E+03
Sr-90	8.07E+04
Y-90	8.07E+04
Ru-106	1.76E+04
Cs-134	5.05E+04
Cs-137	1.10E+05
Pu-238	7.98E+03
Pu-239	2.61E+02
Pu-240	3.99E+02

Table 3-4 Radionuclide Inventory Parameter Values (Continued)

Element	Radionuclide Inventory (Curies)[(a)]
Am-241	1.12E+03
Pu-241	1.03E+05
Cm-244	1.42E+04

Scientific notation is denoted by E followed by the exponent. For example, 5.02 × 10-2 is indicated by 5.02E-02.
Am = americium; Cm = curium; Co = cobalt; Cs = cesium; Kr = krypton; Sr = strontium; Ru = ruthenium; Pu = plutonium; Y = yttrium.
(a) Radionuclide inventories are based the highest curie value for each radionuclide (see Appendix A) based on the NAC International-Legal Weight Truck (NAC-LWT) package capacity of one pressurized water reactor fuel assembly or two boiling water reactor fuel assemblies adjusted to 0.5 MTU.

3.7 Transportation Evaluation

The NRC staff performed an independent evaluation of the environmental impacts as a result of the deployment and use of ATF. By applying the information and data from the previous transportation sections of this NUREG in the NRC-RADTRAN code, this section addresses the environmental impacts from normal operating conditions (radiological impacts) and accident conditions (radiological and nonradiological impacts) resulting from the shipment of unirradiated fuel, and shipment of spent fuel to a permanent geologic repository. The Yucca Mountain site has been used in past NRC environmental reviews as a surrogate geologic repository and is also used as the destination site for spent ATF shipments for this evaluation. To address all forms of waste from the deployment and use of ATF, a discussion of the environmental impacts from shipments of LLRW to offsite disposal facilities during operations is qualitatively assessed based on the history of LLRW shipments.

Radiation exposures at some level due to unirradiated and spent ATF shipments would occur to the following individuals: (1) persons residing along the transportation corridors between the originating and the destination sites; (2) persons in vehicles traveling on the same route as a spent ATF shipment; (3) persons at vehicle stops for refueling, rest, and vehicle inspections; and (4) transportation crew workers. The last group, transportation crew workers, would be radiologically trained and qualified personnel under the 10 CFR Part 20 (TN283) regulations for occupational exposures.

The principal analysis is for the shipment of spent ATF due to its higher potential to have radiological impacts. Thus, the impacts of each of the six nuclear power sites are assessed for spent ATF shipments. Due to the difference in transport package release fractions between BWR and PWR fuel assemblies (see Table 7.31 of Sprung et al. 2000-TN222), both were assessed from each NPP site regardless of which type of NPP was at a site. Due to the lower radiological content in unirradiated ATF shipments (as compared to irradiated ATF shipments), only one shipment case for unirradiated shipments with the longest distance was evaluated as a sensitivity case. As previously mentioned, this case is a shipment from Framatome FFF outside of Richland, Washington, to the Turkey Point site, with a distance of approximately 3,187 mi, or 5,129 km.

3.7.1 Shipments of Low-Level Radioactive Waste

As discussed in Section 3.11.1.1 of the 2013 License Renewal GEIS, LLRW shipments from NPPs to disposal facilities or waste processing centers and from waste processing centers to disposal facilities are generally made by truck. This section of the License Renewal GEIS also discusses the annual quantities of LLRW generated at the NPPs. The quantity of LLRW shipped from NPPs varies from year to year depending on the number of maintenance activities undertaken and the number of unusual occurrences taking place in that year. On average, the volume of LLRW generated at a PWR is approximately 10,600 ft^3 (300 m^3) per year (Table 6.6 in NRC 1996-TN288). The annual volume of LLRW generated at a BWR is approximately twice the values indicated for a PWR. The total volume of LLRW from all sources shipped to the various disposal sites has also varied over time. For the period from 2015 to 2019, the total volume from all sources ranged from about 36,600 m^3 to 144,200 m^3, with a median value of approximately 120,300 m^3 (DOE 2020-TN6669). Thus, the average quantity from an NPP, namely 300 to 600 m^3, would be a small fraction of the annual amount of LLRW shipped nation-wide.

The deployment and use of ATF with increased enrichment and higher burnup levels would not significantly change the annual quantity of LLRW generated at NPPs. The levels of fission products and activated corrosion products present in the primary coolant (the principal source for radiological contamination from maintenance activities) are controlled and monitored routinely. For example, technical specifications for fuel performance and limiting primary to secondary water leakage would be required for ATF as for the current LWR fuels. The other comprehensive regulatory controls that are in place, such as under 10 CFR Part 20 (e.g., 10 CFR 20.1101(b) [TN283] for maintaining radiation exposure as low as is reasonably achievable from all radiation sources, including LLRW, and 10 CFR 20.1406 on minimization of contamination), would ensure that the radiological impacts from LLRW generated from deploying ATF would remain within regulatory limits. Additionally, licensees are required in the Annual Radioactive Effluent Release Report to disclose their radioactive effluents and their impacts on the environment on an annual basis, which includes the impacts from solid radioactive waste. Therefore, the NRC regulations would ensure that the radiological impacts from LLRW generated from deploying ATF would remain small.

In NUREG-0170, “Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes” (NRC 1977-TN417, NRC 1977-TN6497), the NRC evaluated the shipment of radioactive material, including shipments of unirradiated fuel, SNF, and radioactive waste to and from NPPs. The NRC concluded in NUREG-0170 that the average radiation dose to the population at risk from normal transportation is a small fraction of the limits for members of the general public from all sources of radiation other than natural and medical sources (i.e., 10 CFR 20.1301 TN283) and is a small fraction of the natural background dose (NRC 1977-TN417). In addition, the NRC determined that the radiological risk from accidents in transportation is small, amounting to about 0.5 percent of the normal transportation risk on an annual basis. The NRC also determined in NUREG-0170 that the environmental impacts of normal transportation of radioactive materials and the risks attendant to accidents involving radioactive material shipments are sufficiently small to allow continued shipments by all modes. The doses from radioactive waste accidents were negligible when compared to the doses from accidents involving spent fuel shipments. Previous LWR early site permit and combined license (COL) environmental analyses of the nonradiological impacts from accidents involving the transportation of LLRW (injuries and death from physical collisions involving truck LLRW shipments) have shown the risks to be low with small environmental impacts. Since ATF-generated LLRW would not be significantly different than LLRW associated with current LWR fuel, the LLRW impacts assessed for these LWRs would also bound accidents involving ATF-generated LLRW.

Therefore, based on the amount of LLRW shipped annually from a NPP, which is a small fraction of all LLRW shipments and the low risks and environmental impacts from such shipments, the NRC staff finds that LLRW shipment impacts due to deployment and use of ATF with increased enrichment and higher burnup levels would not significantly contribute to the impacts listed in Table S-4.

3.7.2 Shipments of Unirradiated Accident Tolerant Fuel

Instead of determining the unirradiated fuel transportation impacts from each ATF fabrication facility to each of the six plant sites, the staff analyzed a single route with the longest travel distance as a representative route for all NPPs. The selected route is from Framatome FFF outside of Richland, Washington, to the Turkey Point site, a distance of approximately 3,187 mi (5,129 kilometers [km]). As previously mentioned, all unirradiated ATF shipments are assumed to be by truck using Traveller packages for PWR fuel and RAJ-II packages for BWR fuel. Radiation exposures at some level would occur to the four groups of individuals previously discussed.

One of the key inputs to the analysis in WASH-1238 (AEC 1972-TN22) for the reference LWR unirradiated fuel shipments is that the radiation dose rate at 3 ft from the transport vehicle is about 0.1 mrem/hr. The NRC staff also used this dose rate in its analysis of the unirradiated ATF shipments. This chosen dose rate is reasonable because the ATF materials would be low-dose-rate uranium radionuclides and would be packaged similarly to those described in WASH-1238 (i.e., inside a package that provides limited radiation shielding).

Radiological impacts of normal conditions as well as nonradiological transportation accident impacts were evaluated using route information from WebTRAGIS and other input values for the NRC-RADTRAN code (see Appendix A, Appendix C, and Appendix D) to determine the impacts based on a per shipment basis. The amount of radioactivity contained in unirradiated ATF is significantly less than that for spent ATF such that any radiological release of unirradiated ATF under accident conditions does not have the potential for a significant health effect. Thus, spent ATF transportation accident radiological impacts would bound any unirradiated ATF transportation accident radiological impacts. These single shipment results were then adjusted to annual impacts based on the number of expected annual shipments to support core reloads (one-third and half of the fuel assemblies in a core) on a 2-year refueling cycle. The overall normal condition radiological impacts on populations are presented in Table 3-5 and Table 3-6; unirradiated fuel accident impacts are presented in Table 3-7 and Table 3-8. The complete table of unirradiated ATF shipment impacts (e.g., impacts per shipment and total impacts) is provided in Appendix E. Nonradiological impacts are based on the annual shipments of unirradiated ATF as well as the return trip of the empty packages. Shipments of Spent Accident Tolerant Fuel.

In this section, the NRC staff evaluates the environmental effects of spent ATF shipments at higher burnup levels than previously assessed in other NRC studies. The evaluation is conducted in a manner similar to past studies, by analyzing the radiological impacts from normal conditions, or incident-free, transportation of spent ATF and for transportation accidents—both the radiological impacts from potential releases of radioactive material and the nonradiological impacts from vehicle accidents. This analysis also addresses sensitivity cases by assessing impacts by examining rail shipments and potential effects due to higher radiological material release fractions from the physical effects of higher burnup levels on the fuel pin cladding and the uranium fuel pellets (see Section B.1 of Appendix B).

Table 3-5 Total Annual Shipment Radiological Impacts for Unirradiated Accident Tolerant Fuel for One-Third-Core Reload

Site (Reactor Fuel Type)	One-Way Shipping Distance (miles)	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Cumulative Public Dose (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition[(a)]	—	<1 per day	4.0E+00	—	—	3.0E+00
Turkey Point (BWR)	3,187	4	5.07E-02	2.72E-01	1.10E-03	2.73E-01
Turkey Point (PWR)[(b)]	3,187	3	3.80E-02	2.04E-01	8.25E-04	2.05E-01

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
BWR = boiling water reactor; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Cumulative public dose in Table S-4 is related to the combined impacts of the transportation of fuel (unirradiated and spent) and solid radioactive waste.
(b) Denotes the reactor type at the site location under the current NRC license.

Table 3-6 Total Annual Shipment Radiological Impacts for Unirradiated Accident Tolerant Fuel for Half-Core Reload

Site (Reactor Fuel Type)	One-Way Shipping Distance (miles)	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Cumulative Public Dose (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition[(a)]	—	<1 per day	4.0E+00	—	—	3.0E+00
Turkey Point (BWR)	3,187	6	7.60E-02	4.07E-01	1.65E-03	4.09E-01
Turkey Point (PWR)[(b)]	3,187	5	6.34E-02	3.40E-01	1.37E-03	3.41E-01

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
BWR = boiling water reactor; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Cumulative public dose in Table S-4 is related to the combined impacts of the transportation of fuel (unirradiated and spent) and solid radioactive waste.
(b) Denotes the reactor type at the site location under the current NRC license.

Table 3-7 Total Annual Unirradiated Fuel Accident Impacts for One-Third-Core Reload

Site (Reactor Fuel Type)	One-Way Shipping Distance (miles)	No. of Normalized Annual Truck Shipments	Total Accident Risks	Total Fatalities Risk	Total Injuries Risk
10 CFR 51.52 (TN250), Table S-4 Condition	—	—	—	0.01	0.1
Turkey Point (Unirradiated Accident Tolerant Fuel – BWR)	3,187	4	1.10E-2	3.71E-4	4.27E-3
Turkey Point (Unirradiated Accident Tolerant Fuel – PWR)[(a)]	3,187	3	8.28E-3	2.78E-4	3.20E-3

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
BWR = boiling water reactor; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table 3-8 Total Annual Unirradiated Fuel Nonradiological Accident Impacts for Half-Core Reload

Site (Reactor Fuel Type)	One-Way Shipping Distance (miles)	No. of Normalized Annual Truck Shipments	Total Accident Risks	Total Fatalities Risk	Total Injuries Risk
10 CFR 51.52 (TN250), Table S-4 Condition	—	—	—	0.01	0.1
Turkey Point (Unirradiated Accident Tolerant Fuel – BWR)	3,187	6	1.66E-02	5.57E-04	6.41E-03
Turkey Point (Unirradiated Accident Tolerant Fuel – PWR)[(a)]	3,187	5	1.38E-02	4.64E-04	5.34E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
BWR = boiling water reactor; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

The NRC staff's evaluation is based on shipments of spent ATF by legal weight trucks (capacity of up to 80,000 lb) in shipping casks that have characteristics similar to currently available transport packages (i.e., massive, heavily shielded, cylindrical metal pressure vessels). Due to the large size and weight of spent fuel transport packages, each shipment is assumed to consist of a single transport package loaded on a modified trailer. These assumptions are consistent with those made in the evaluation of the environmental impacts of transportation of spent fuel in WASH-1238 (AEC 1972-TN22), Addendum 1 to the License Renewal GEIS (NRC 1999-TN289) and Ramsdell et al. (2001-TN4545). The truck transport assumptions are conservative because the alternative transportation methods involve rail or heavy-haul truck transportation of larger transport packages with large capacities for the number of spent fuel assemblies, which would result in significantly fewer shipments than the overall number of spent fuel shipments by truck (NRC 1999-TN289). Therefore, rail or heavy-haul truck transportation are expected to have lower associated impacts.

3.7.2.1 Impacts of Normal Conditions

Under normal conditions, impacts from spent ATF shipments would be from the regulated levels of radiation that penetrate the package's shielding. Radiation exposures at some level would occur to the four groups of individuals previously discussed. Due to the nature of the spent ATF, the transportation crew workers would be radiologically trained and qualified where the 10 CFR Part 20 (TN283) regulations for occupational exposures would apply.

This evaluation assumes that individual transportation crew member doses are limited to 2 rem/yr, which is the DOE administrative control level presented in DOE-STD-1098-99, DOE Standard, Radiological Control, Chapter 2, Article 211 (DOE 2005-TN1235). This dose limit is anticipated to apply to spent ATF shipments to a disposal site because DOE would ultimately take title to the spent fuel at the reactor site and be responsible for conducting the SNF shipments per the Nuclear Waste Policy Act. Such a dose limit would also be reasonable for non-DOE shipments of spent ATF to an offsite storage facility (i.e., a CISF). As cited for unirradiated ATF shipments, the input parameter values used in the NRC-RADTRAN calculations for a single shipment are provided in Appendix D and subsequent total impacts applying the NRC-RADTRAN calculations for each site and for each type of reactor fuel (BWR and PWR) based on the number of annual shipments are provided in Appendix E. The radiological impacts due to normal transportation conditions from the various sites on an annual basis are shown in Table 3-5 through Table 3-10.

Results from WASH-1238 (AEC 1972-TN22) and the prior transportation environmental evaluation at higher burnups, namely that of Ramsdell et al. (2001-TN4545), are also provided to aid in assessing the updated evaluation. The general trend for all sites for normal conditions worker doses was that Table S-4 bounds the results for ATF shipments, whether for unirradiated or spent ATF with enrichments such as with a maximum 8 wt% U-235 and assembly averaged burnup levels up to 80 GWd/MTU. It is also clear that the ATF shipment results are strongly tied to the shipment distance, the number of annual shipments, and the size of a route's population. The effects of the routes' populations are expressly shown in Table 3-9 and Table 3-10. The sites that have a significantly greater population along a route (i.e., Brunswick, Millstone, and Turkey Point) have a cumulative dose, especially for BWR fuels, higher than the 3 person-rem per year specified in the Table S-4. However, these results are only marginally higher than the 3 person-rem of Table S-4 and are not significant given the individual doses considered. For example, when the average individual dose for the route population is assessed, the values as shown in Table 3-11, and Table 3-12 are well below 1 mrem per year and within the Table S-4 range of doses to exposed individuals. Moreover, the average individual doses are also a small fraction of the expected annual natural background radiation dose of 310 mrem/yr for both sizes of core reload. While the cumulative public doses for both sizes of core reload are greater than the 3 person-rem of Table S-4, they are not a significant environmental impact due to being a very small fraction of the expected background radiation dose (e.g., for the Columbia route with a total population of approximately 60,286 results in an annual total natural background radiation dose of approximately 18,689 person-rem/yr versus a 4.86 person-rem/yr for shipment of a BWR half-core reload). For the transportation crews, the transportation evaluation demonstrates that their cumulative doses would be bounded by the 4 person-rem of Table S-4 for all sites. Therefore, this transportation evaluation of the effects of ATF shipments of up to 8 wt% U-235 and 80 GWd/MTU demonstrates that Table S-4 is still bounding for normal conditions of ATF transport for the assumptions and conditions applied.

Table 3-9 Total Annual Shipment Radiological Impacts for Spent Accident Tolerant Fuel for One-Third-Core Reload

Site (Reactor Fuel Type)	One Way Miles per Shipment	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Cumulative Public Dose (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition[(a)]	—	<1 per day	4.0E+00	—	—	3.0E+00
Brunswick (BWR)[(b)]	2,475	52	2.56E+00	7.14E+00	4.00E-01	7.54E+00
Brunswick (PWR)	2,475	30	1.48E+00	4.12E+00	2.31E-01	4.35E+00
Columbia (BWR)[(b)]	908	52	9.51E-01	3.19E+00	5.01E-02	3.24E+00
Columbia (PWR)	908	30	5.49E-01	1.84E+00	2.89E-02	1.87E+00
Dresden (BWR)[(b)]	1,843	52	1.87E+00	4.46E+00	1.63E-01	4.62E+00
Dresden (PWR)	1,843	30	1.08E+00	2.57E+00	9.38E-02	2.67E+00
Fermi (BWR)[(b)]	2,131	52	2.21E+00	4.92E+00	2.52E-01	5.17E+00
Fermi (PWR)	2,131	30	1.27E+00	2.84E+00	1.45E-01	2.99E+00
Millstone (BWR)	2,770	52	2.92E+00	7.61E+00	4.25E-01	8.04E+00
Millstone (PWR)[(b)]	2,770	30	1.68E+00	4.39E+00	2.45E-01	4.64E+00
Turkey Point (BWR)	2,642	52	2.73E+00	6.56E+00	4.49E-01	7.01E+00
Turkey Point (PWR)[(b)]	2,642	30	1.58E+00	3.78E+00	2.59E-01	4.04E+00
NUREG/CR-6703 (BWR-NE) (75 GWd/MTU)[(c)]	2,637	17.5	0.39	1.40	2.76	N/A
NUREG/CR-6703 (PWR-SE) (75 GWd/MTU)[(c)]	2,832	14.8	0.34	1.22	2.57	N/A

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; N/A = not applicable; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Cumulative public dose in Table S-4 is related to the combined impacts of the transportation of fuel (unirradiated and spent) and solid radioactive waste.
(b) Denotes the reactor type at the site location under the current NRC license.
(c) NUREG/CR-6703 results for the highest burnup level, 75 GWd/MTU, are for the highest doses from Table 7.4, 7.6, and 7.7 for the Southeast and Northeast regions (i.e., Turkey Point and Millstone) based on four BWR and two PWR spent fuel assemblies per shipment (Ramsdell et al. 2001-TN4545).

Table 3-10 Total Annual Shipment Radiological Impacts for Spent Accident Tolerant Fuel for Half-Core Reload

Site (Reactor Fuel Type)	One Way Miles per Shipment	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Cumulative Public Dose (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition[(a)]	—	<1 per day	4.0E+00	—	—	3.0E+00
Brunswick (BWR)[(b)]	2,475	78	3.85E+00	1.07E+01	5.99E-01	1.13E+01
Brunswick (PWR)	2,475	45	2.22E+00	6.18E+00	3.46E-01	6.53E+00
Columbia (BWR)[(b)]	908	78	1.43E+00	4.79E+00	7.52E-02	4.86E+00
Columbia (PWR)	908	45	8.23E-01	2.76E+00	4.34E-02	2.81E+00
Dresden (BWR)[(b)]	1,843	78	2.81E+00	6.69E+00	2.44E-01	6.94E+00
Dresden (PWR)	1,843	45	1.62E+00	3.86E+00	1.41E-01	4.00E+00
Fermi (BWR)[(b)]	2,131	78	3.31E+00	7.38E+00	3.78E-01	7.76E+00
Fermi (PWR)	2,131	45	1.91E+00	4.26E+00	2.18E-01	4.48E+00
Millstone (BWR)	2,770	78	4.38E+00	1.14E+01	6.37E-01	1.21E+01
Millstone (PWR)[(b)]	2,770	45	2.53E+00	6.59E+00	3.68E-01	6.95E+00
Turkey Point (BWR)	2,642	78	4.10E+00	9.83E+00	6.73E-01	1.05E+01
Turkey Point (PWR)[(b)]	2,642	45	2.37E+00	5.67E+00	3.88E-01	6.06E+00
NUREG/CR-6703 (BWR-NE) (75 GWd/MTU)[(c)]	2,637	17.5	0.39	1.40	2.76	N/A
NUREG/CR-6703 (PWR-SE) (75 GWd/MTU)[(c)]	2,832	14.8	0.34	1.22	2.57	N/A

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; N/A = not applicable; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Cumulative public dose in Table S-4 is related to the combined impacts of the transportation of fuel (unirradiated and spent) and solid radioactive waste.
(b) Denotes the reactor type at the site location under the current NRC license.
(c) NUREG/CR-6703 results for the highest burnup level, 75 GWd/MTU, are for the highest doses from Table 7.4, 7.6, and 7.7 for the Southeast and Northeast regions (i.e., Turkey Point and Millstone) based on four BWR and two PWR spent fuel assemblies per shipment (Ramsdell et al. 2001-TN4545).

Table 3-11 Average Annual Individual Radiological Dose to Total, Along Route, and Onlooker Populations for One-Third-Core Reload

Site/Reactor Type	Total Population Along the Route	Individual Population Averaged Annual Dose (mrem)	Along Route Population	Along Route Population Average Annual Dose (mrem)	Onlooker Population	Onlooker Population Average Annual Dose (mrem)
10 CFR 51.52 (TN250), Table S-4 Condition[(a)]	601,100	—	600,000	0.0001–0.06	1,100	0.003–1.3
Brunswick (BWR)[(b)]	1,022,499	0.00738	923,789	0.00043	98,710	0.07238
Brunswick (PWR)	1,022,499	0.00426	923,789	0.00025	98,710	0.04176
Columbia (BWR)[(b)]	94,344	0.03436	60,286	0.00083	34,058	0.09371
Columbia (PWR)	94,344	0.01982	60,286	0.00048	34,058	0.05406
Dresden (BWR)[(b)]	461,805	0.01001	406,886	0.00040	54,919	0.08124
Dresden (PWR)	461,805	0.00578	406,886	0.00023	54,919	0.04687
Fermi (BWR)[(b)]	658,906	0.00785	586,871	0.00043	72,035	0.06834
Fermi (PWR)	658,906	0.00453	586,871	0.00025	72,035	0.03943
Millstone (BWR)	1,177,724	0.00682	1,063,230	0.00040	114,494	0.06647
Millstone (PWR)[(b)]	1,177,724	0.00394	1,063,230	0.00023	114,494	0.03835
Turkey Point (BWR)	1,468,716	0.00477	1,361,975	0.00033	106,741	0.06143
Turkey Point (PWR)[(b)]	1,468,716	0.00275	1,361,975	0.00019	106,741	0.03544

Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR= pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.

(a) From Summary Table S-4 in NUREG-75/038 (NRC 1975-TN216).

(b) Denotes the reactor type at the site location under the current NRC license.

Table 3-12 Average Annual Individual Radiological Dose to Total, Along Route, and Onlooker Populations for Half-Core Reload

Site/Reactor Type	Total Population Along the Route	Individual Population Averaged Annual Dose (mrem)	Along Route Population	Along Route Population Average Annual Dose (mrem)	Onlooker Population	Onlooker Population Average Annual Dose (mrem)
10 CFR 51.52 (TN250), Table S-4 Condition[a]	601,100	—	600,000	0.0001–0.06	1,100	0.003–1.3
Brunswick (BWR)[b]	1,022,499	0.01107	923,789	0.00065	98,710	0.10858
Brunswick (PWR)	1,022,499	0.00639	923,789	0.00037	98,710	0.06264
Columbia (BWR)[b]	94,344	0.05154	60,286	0.00125	34,058	0.14056
Columbia (PWR)	94,344	0.02973	60,286	0.00072	34,058	0.08109
Dresden (BWR)[b]	461,805	0.01502	406,886	0.00060	54,919	0.12185
Dresden (PWR)	461,805	0.00866	406,886	0.00035	54,919	0.07030
Fermi (BWR)[b]	658,906	0.01178	586,871	0.00064	72,035	0.10251
Fermi (PWR)	658,906	0.00680	586,871	0.00037	72,035	0.05914
Millstone (BWR)	1,177,724	0.01023	1,063,230	0.00060	114,494	0.09971
Millstone (PWR)[b]	1,177,724	0.00590	1,063,230	0.00035	114,494	0.05752
Turkey Point (BWR)	1,468,716	0.00715	1,361,975	0.00049	106,741	0.09214
Turkey Point (PWR)[b]	1,468,716	0.00413	1,361,975	0.00029	106,741	0.05316

Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR= pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.

(a) From Summary Table S-4 in NUREG-75/038 (NRC 1975-TN216).

(b) Denotes the reactor type at the site location under the current NRC license.

3.7.2.2 Accident Impacts

As discussed previously, the NRC staff used the NRC-RADTRAN computer code to estimate the impacts of transportation accidents involving spent fuel shipments. NRC-RADTRAN considers a spectrum of postulated transportation accidents, ranging from those with high frequencies and low consequences (e.g., “fender benders”) to those with low frequencies and high consequences (i.e., accidents in which the shipping container is exposed to severe mechanical and thermal conditions). The radionuclide inventories are important parameters in the calculation of accident risks. The radionuclide inventory used in this evaluation is discussed in Appendix A.

Robust shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR Part 71 (TN301). Spent fuel shipping casks must be certified Type B packaging systems, meaning they must withstand a series of severe postulated accident conditions with essentially no loss of containment or shielding capability. These casks also are designed with fissile material controls to ensure that the spent fuel remains subcritical under normal and accident conditions. According to Sprung et al. (2000-TN222), the probability of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). For this evaluation, the NRC staff considered that transport packages approved for the transportation of the spent ATF would provide equivalent mechanical and thermal protection of the spent fuel cargo as previously analyzed by Sprung et al. (2000-TN222).

Accident frequencies are calculated in NRC-RADTRAN using user-specified accident rates and conditional shipping cask failure probabilities. As discussed in Section 3.5.4, State-specific accident rates used in the NRC-RADTRAN calculations were extracted from a FMCSA database and are provided in Appendix E. The release of radioactive material in the NRC-RADTRAN calculations is based on the severity levels and package release fractions as discussed in Section 3.5.4 and noted in Appendix D. The nonradiological vehicle accident fatality and injury rates by State are also from DOT databases as provided in Appendix E and were used to generate the annual nonradiological accident fatality and injury risks for shipments to each site.

Overall, the results shown in Table 3-13 through Table 3-16 demonstrate the low risks for both radiological and nonradiological accident risks from unirradiated and spent ATF shipments at a maximum of 8 wt% U-235 and up to 80 GWd/MTU. This is consistent with the conclusion of WASH-1238 (AEC 1972-TN22) and NUREG-75/038 (NRC 1975-TN216) codified in Table S-4 that the transportation radiological accident impacts would be small. The results of this study are also lower than the previous evaluation provided by Ramsdell et al. (2001-TN4545). This is principally due to the differences in assessing accidents between RADTRAN 4 and NRC-RADTRAN along with differences in the values and assumptions applied by Ramsdell et al. (2001-TN4545). For example, the release fractions used by Ramsdell et al. (2001-TN4545) are different from those developed by Sprung et al. (2000-TN222) applied in this study.

Another item that appears in the results is the difference between BWR and PWR radiological and nonradiological accident impacts. Radiological PWR risks are greater than the BWR risks even though there are more shipments per year of spent BWR ATF. This radiological accident difference is attributed to the differences in release fraction provided in Table 7.31 of Sprung et al. (2000-TN222) for the steel-lead-steel truck package. For example, under Case 2 of this table for all five chemical categories, the BWR release fractions are less than the PWR release fractions. There are also cases where the BWR release fractions are greater than the PWR release fraction. Overall, there are more cases with BWR values less than PWR values to yield the results given in Table 3-13 and Table 3-14. The nonradiological BWR and PWR accident impact differences are the opposite (i.e., BWR impacts are greater than PWR impacts) and driven by the number of annual shipments. Vehicle accident rates applied in this study are based on commercial freight truck accident rates and the same values were applied to both BWR and PWR shipments. Thus, with BWRs having more annual shipments, their nonradiological impacts will be greater than PWR annual shipments.

Table 3-13 Radiological Accident Impacts of Spent Accident Tolerant Fuel for One-Third-Core Reload

Site (Reactor Fuel Type)	Total Miles per Shipment	No. of Normalized Annual Shipments	Total Accident Risk (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition	—	<1 per day	—
Brunswick (BWR)[(b)]	2,475	52	4.87E-06
Brunswick (PWR)	2,475	30	9.57E-06
Columbia (BWR)[(b)]	908	52	1.78E-07
Columbia (PWR)	908	30	3.48E-07
Dresden (BWR)[(b)]	1,843	52	1.92E-06
Dresden (PWR)	1,843	30	3.78E-06
Fermi (BWR)[(b)]	2,131	52	3.14E-06
Fermi (PWR)	2,131	30	6.18E-06
Millstone (BWR)	2,770	52	8.11E-06
Millstone (PWR)[(b)]	2,770	30	1.59E-05
Turkey Point (BWR)	2,642	52	1.00E-05
Turkey Point (PWR)[(b)]	2,642	30	1.97E-05
NUREG/CR-6703 (2001-TN4545) (BWR-NE) (75 GWd/MTU)[(a)]	2,637	17.5	0.041
NUREG/CR-6703 (2001-TN4545) (PWR-SE) (75 GWd/MTU)[(a)]	2,832	14.8	0.064

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Ramsdell et al. (2001-TN4545).
(b) Denotes the reactor type at the site location under the current NRC license.

Table 3-14 Radiological Accident Impacts of Spent Accident Tolerant Fuel for Half-Core Reload

Site (Reactor Fuel Type)	Total Miles per Shipment	No. of Normalized Annual Shipments	Total Accident Risk (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition	—	<1 per day	—
Brunswick (BWR)[(b)]	2,475	78	7.30E-06
Brunswick (PWR)	2,475	45	1.44E-05
Columbia (BWR)[(b)]	908	78	2.67E-07
Columbia (PWR)	908	45	5.22E-07
Dresden (BWR)[(b)]	1,843	78	2.88E-06
Dresden (PWR)	1,843	45	5.67E-06
Fermi (BWR)[(b)]	2,131	78	4.71E-06
Fermi (PWR)	2,131	45	9.27E-06
Millstone (BWR)	2,770	78	1.22E-05
Millstone (PWR)[(b)]	2,770	45	2.39E-05
Turkey Point (BWR)	2,642	78	1.51E-05
Turkey Point (PWR)[(b)]	2,642	45	2.96E-05
NUREG/CR-6703 (2001-TN4545) (BWR-NE) (75 GWd/MTU)[(a)]	2,637	17.5	0.041
NUREG/CR-6703 (2001-TN4545) (PWR-SE) (75 GWd/MTU)[(a)]	2,832	14.8	0.064

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR = pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Ramsdell et al. (2001-TN4545).
(b) Denotes the reactor type at the site location under the current NRC license.

3.7.3 Sensitivity Analysis

As sensitivity cases, the NRC staff examines the environmental effects if spent ATF is transported by rail instead of by truck and reassesses the release of radioactive material resulting from the burnup levels higher than those previously evaluated by Sprung et al. (2000-TN222).

3.7.3.1 Rail Shipment Sensitivity Analysis

The rationale for conducting a rail sensitivity case stems from the potential for rail transport packages to hold significantly more spent ATF assemblies than other forms of transportation. There are indications that the industry and DOE would most likely use this transportation pathway over others due to several factors such as overall costs for a shipping campaign or rail transport package compatibility with dry cask storage systems, among other factors. It is not expected that rail transportation will be chosen based solely on reducing the number of shipments required to move the same number of spent ATF assemblies.

As discussed in Section 3.6.2 of this study, the NAC-STC rail transport package was selected for the rail shipments evaluation. Using this package results in annual shipments of PWR spent

ATF of approximately 1.25 shipments per year. Prior SNF shipment evaluations have also assessed rail transport. These include the Yucca Mountain EIS (DOE 2002-TN1236), NUREG-2125 (NRC 2014-TN3231), and both CISF EISs (NRC 2020-TN6499, NRC 2020-TN6498). Applying this information and the number of assemblies the NAC-STC can hold, the environmental impacts from each of the six NPP sites are shown in Table 3-17 and Table 3-18. These results are significantly less than the PWR spent ATF truck shipment impacts shown in Table 3-5, Table 3-6, Table 3-13, and Table 3-14, and the environmental impacts of Table S-4.

3.7.3.2 Release Fractions Sensitivity Analysis

The previous study of the environmental impacts of spent fuel transportation by Ramsdell et al. (2001-TN4545) indicated there are no significant adverse environmental impacts associated with extending peak-rod fuel burnup to 62 GWd/MTU. The factor limiting this conclusion as presented by Ramsdell et al. (2001-TN4545) to 62 GWd/MTU is uncertainty in changes in the gap-release fraction associated with increasing fuel burnup. Also, Ramsdell et al. (2001-TN4545) did not have access to the release fractions generated by Sprung et al. (2000-TN222) for use in their RADTRAN4 transportation calculations. Additionally, the maximum burnup levels applied by Sprung et al. (2000-TN222) did not go above 60 GWd/MTU. Thus, the question arises whether the transportation accident impacts could significantly change at burnup levels above the 60 GWd/MTU of Sprung et al. (2000-TN222) given that higher burnup levels could also affect the release fractions due to cladding embrittlement, fuel fragmentation, and diffusional release of fission products.

PNNL was contracted to examine and assess the potential effects on the transport release fractions under burnup levels greater than 60 GWd/MTU for BWR and PWR spent fuel assemblies. The discussion and results of this examination of release fractions at higher burnups can be found in Appendix B of this study. The release fractions developed for 72 (Table B-9 and Table B-10) and 85 GWd/MTU (Table B-12 and Table B-13) were applied to the case of shipping spent ATF by truck from the Turkey Point site. Truck shipments were selected for this sensitivity analysis based on the larger number of shipments by truck versus rail with the resulting larger accident risks from the truck transport being several orders of magnitude greater than those of rail transport. The resulting truck transport accident risks are shown in Table 3-19 and Table 3-20, along with the previous results for Turkey Point. The normal condition risks are provided as a benchmark to show consistency between the calculations and to demonstrate these impacts are independent of the accident impacts.

An approximate two orders of magnitude change in risk was observed with the revised accident release fractions for the two sensitivity analysis cases of higher burnup from the conditions in Sprung et al. (2000-TN222). This increase in risk is principally attributed to the particulate release fraction. There is an increase in the volume of the pellet that has fragmented (i.e., transformed to a higher burnup rim structure (fragmentation) that is available as particulate release); the fragmented volume increases to 20 percent at 85 GWd/MTU. However, while the increase in accident risk is noticeable, the accident risk values for such higher burnup are still not significant.

Table 3-15 Nonradiological Accident Impacts of Spent Accident Tolerant Fuel for One-Third-Core Reload

Site	Normalized Annual Truck Shipments	One-Way Shipping Distance (miles)	Total Accident Risks	Total Fatalities Risk	Total Injuries Risk
Brunswick (BWR)[a]	52	2,475	1.15E-01	4.64E-03	4.80E-02
Brunswick (PWR)	30	2,475	6.66E-02	2.68E-03	2.77E-02
Columbia (BWR)[a]	52	908	3.11E-02	1.59E-03	1.21E-02
Columbia (PWR)	30	908	1.79E-02	9.18E-04	6.96E-03
Dresden (BWR)[a]	52	1,843	7.20E-02	2.30E-03	2.49E-02
Dresden (PWR)	30	1,843	4.15E-02	1.33E-03	1.43E-02
Fermi (BWR)[a]	52	2,131	9.10E-02	2.81E-03	3.14E-02
Fermi (PWR)	30	2,131	5.25E-02	1.62E-03	1.81E-02
Millstone (BWR)	52	2,770	1.40E-01	3.93E-03	5.27E-02
Millstone (PWR)[a]	30	2,770	8.10E-02	2.27E-03	3.04E-02
Turkey Point (BWR)	52	2,642	1.27E-01	5.16E-03	6.20E-02
Turkey Point (PWR)[a]	30	2,642	7.32E-02	2.98E-03	3.58E-02
10 CFR 51.52 (TN250), Table S-4 Condition	—	—	—	0.01	0.1

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR= pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table 3-16 Nonradiological Accident Impacts of Spent Accident Tolerant Fuel for Half-Core Reload

Site	Normalized Annual Truck Shipments	One-Way Shipping Distance (miles)	Total Accident Risks	Total Fatalities Risk	Total Injuries Risk
Brunswick (BWR)[(a)]	78	2,475	1.73E-01	6.96E-03	7.21E-02
Brunswick (PWR)	45	2,475	9.99E-02	4.01E-03	4.16E-02
Columbia (BWR)[(a)]	78	908	4.66E-02	2.39E-03	1.81E-02
Columbia (PWR)	45	908	2.69E-02	1.38E-03	1.04E-02
Dresden (BWR)[(a)]	78	1,843	1.08E-01	3.45E-03	3.73E-02
Dresden (PWR)	45	1,843	6.23E-02	1.99E-03	2.15E-02
Fermi (BWR)[(a)]	78	2,131	1.37E-01	4.21E-03	4.71E-02
Fermi (PWR)	45	2,131	7.88E-02	2.43E-03	2.72E-02
Millstone (BWR)	78	2,770	2.11E-01	5.90E-03	7.91E-02
Millstone (PWR)[(a)]	45	2,770	1.22E-01	3.40E-03	4.56E-02
Turkey Point (BWR)	78	2,642	1.90E-01	7.74E-03	9.30E-02
Turkey Point (PWR)[(a)]	45	2,642	1.10E-01	4.46E-03	5.36E-02
10 CFR 51.52 (TN250), Table S-4 Condition (AEC 1972-TN22)	—	—	—	0.01	0.1

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; BWR = boiling water reactor; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR= pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table 3-17   Sensitivity Rail Transport Impacts for One-Third-Core Reload

Site	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Total Public Dose (person-rem)	Total Accident Population Risk (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition	<3 per month	4.0	—	—	3.0	Small
Brunswick (PWR)	1.25	2.16E-02	8.60E-04	2.05E-02	2.14E-02	7.53E-10
Columbia (PWR)	1.25	1.10E-02	2.65E-04	5.68E-03	5.94E-03	2.17E-10
Dresden (PWR)	1.25	1.52E-02	4.64E-04	9.36E-03	9.83E-03	3.36E-10
Fermi (PWR)	1.25	1.77E-02	6.38E-04	1.55E-02	1.61E-02	6.36E-10
Millstone (PWR)[(a)]	1.25	2.14E-02	8.96E-04	2.38E-02	2.46E-02	9.33E-10
Turkey Point (PWR)[(a)]	1.25	2.34E-02	9.55E-04	2.51E-02	2.61E-02	1.02E-09

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR= pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table 3-18 Sensitivity Rail Transport Impacts for Half-Core Reload

Site	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Total Public Dose (person-rem)	Total Accident Population Risk (person-rem)
10 CFR 51.52 (TN250), Table S-4 Condition	<3 per month	4.0	—	—	3.0	Small
Brunswick (PWR)	1.73	2.99E-02	1.19E-03	2.84E-02	2.96E-02	1.04E-09
Columbia (PWR)	1.73	1.52E-02	3.67E-04	7.85E-03	8.22E-03	3.01E-10
Dresden (PWR)	1.73	2.11E-02	6.42E-04	1.30E-02	1.36E-02	4.65E-10
Fermi (PWR)	1.73	2.44E-02	8.82E-04	2.15E-02	2.23E-02	8.80E-10
Millstone (PWR)[(a)]	1.73	2.96E-02	1.24E-03	3.29E-02	3.41E-02	1.29E-09
Turkey Point (PWR)[(a)]	1.73	3.24E-02	1.32E-03	3.48E-02	3.61E-02	1.41E-09

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station; Columbia = Columbia Generating Station; Dresden = Dresden Generating Station; Fermi = Enrico Fermi Nuclear Generating Station; Millstone = Millstone Nuclear Power Plant; PWR= pressurized water reactor; Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table 3-19 Turkey Point Nuclear Generating Station Truck Transport Sensitivity Analysis Results for 72 and 85 GWd/MTU Based Release Fractions for One-Third-Core Reload

Analysis	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Total Public Dose (person-rem)	Total Accidental Population Risk (person-rem)
BWR — Sprung et al. 2000-TN222	52	2.73E+00	6.56E+00	4.49E-01	7.01E+00	1.00E-05
BWR at 72 GWd/MTU[(a)]	52	2.73E+00	6.56E+00	4.49E-01	7.01E+00	2.05E-03
BWR at 85 GWd/MTU[(a)]	52	2.73E+00	6.56E+00	4.49E-01	7.01E+00	3.97E-03
PWR — Sprung et al. 2000-TN222	30	1.58E+00	3.78E+00	2.59E-01	4.04E+00	1.97E-05
PWR at 72 GWd/MTU[(a)]	30	1.58E+00	3.78E+00	2.59E-01	4.04E+00	1.30E-03
PWR at 85 GWd/MTU	30	1.58E+00	3.78E+00	2.59E-01	4.04E+00	2.49E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
BWR= boiling water reactor; PWR = pressurized water reactor.
(a) The results in this row are based on applying the release fractions from Appendix B to assess higher burnup rates.

Table 3-20 Turkey Point Nuclear Generating Station Truck Transport Sensitivity Analysis Results for 72 and 85 GWd/MTU Based Release Fractions for Half-Core Reload

Analysis	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Total Public Dose (person-rem)	Total Accidental Population Risk (person-rem)
BWR — Sprung et al. 2000-TN222	78	4.10E+00	9.83E+00	6.73E-01	1.05E+01	1.51E-05
BWR at 72 GWd/MTU[(a)]	78	4.10E+00	9.83E+00	6.73E-01	1.05E+01	3.08E-03
BWR at 85 GWd/MTU[(a)]	78	4.10E+00	9.83E+00	6.73E-01	1.05E+01	5.95E-03
PWR — Sprung et al. 2000-TN222	45	2.37E+00	5.67E+00	3.88E-01	6.06E+00	2.96E-05
PWR at 72 GWd/MTU[(a)]	45	2.37E+00	5.67E+00	3.88E-01	6.06E+00	1.95E-03
PWR at 85 GWd/MTU	45	2.37E+00	5.67E+00	3.88E-01	6.06E+00	3.73E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
BWR= boiling water reactor; PWR = pressurized water reactor.
(a) The results in this row are based on applying the release fractions from Appendix B to assess higher burnup rates.

3.8 Accident Tolerant Fuel Transportation Conclusions

The NRC staff performed an independent re-evaluation of the transportation of fuel and waste for the environmental effects expected from the deployment and use of ATF with increased enrichment up to 8 wt% U-235 and burnup levels of up to 80 GWd/MTU. The three principal categories of transportation of fuel and waste, namely for shipments of LLRW, unirradiated ATF, and spent ATF, under normal conditions and accidents are discussed and assessed against the transportation impacts provided in Table S-4 under 10 CFR 51.52(c) (TN250) and past studies as appropriate. The overall conclusion from this transportation evaluation is the radiological and nonradiological environmental risks would still be low for the deployment and use of ATF with the increased enrichment and higher burnup levels for all three categories of radioactive material transportation.

As described in the analysis above, doses to exposed individuals during transport of ATF and waste are low. These radiological doses under normal conditions to exposed individuals for all shipments are within the range of doses to exposed individuals provided in Table S-4 and are a better indicator of ATF shipment impacts than the cumulative public dose. The cumulative public dose values for all six sites are driven by the presence of larger populations along the route considered in this study versus what was analyzed in WASH-1238 (AEC 1972-TN22). It is also worth noting that several of the PWR cumulative public dose values are close to or similar to the Table S-4 value of 3 person-rem so that expanding the time of shipments (1 year for Table S-4 versus 2 years for deployment and use of ATF) resulting in a reduced number of annual shipments is a competing factor relative to the larger populations seen in this analysis along a route. This is also evident from the higher BWR cumulative public doses with the same population along the routes as for the PWR cases with almost double the number of shipments. As another measure of the significance of normal conditions impacts, an average individual dose for the route population is assessed where the values are well below 1 mrem/yr, a small fraction of the average annual natural background radiation exposure of approximately 310 mrem, and within the Table S-4 range of doses to exposed individuals. These results are also based on the transport package that has the least capacity. Applying a transport package with a greater capacity would reduce the number of shipments resulting in a much lower cumulative public dose that would be less than the 3 person-rem of Table S-4, as shown by the rail sensitivity case (e.g., the GA-4 truck spent fuel transport can hold four PWR fuel assemblies, thereby reducing the PWR cumulative public doses by a factor of 4) (NRC 2009-TN8291).

The accident risk results of this study are consistent with past studies, such as those provided in NUREG-2125 (NRC 2014-TN3231), demonstrating the low risks from spent ATF transportation accidents. The radiological risks are much lower in this study than the previous risk results presented by Ramsdell et al. (2001-TN4545) in part due to the changes in the RADTRAN code from Version 4 to Version 6.02, which is the code driver in NRC-RADTRAN, as noted in Section 3.5.1 of this study. Even though there is the potential for higher release fractions at higher burnup levels above 62 GWd/MTU, such higher release fractions, such as at 85 GWd/MTU, still result in relatively low accident risks. The greater risk to a member of the public would be physical harm from the actual vehicle collision with a spent ATF shipment, if such an event happens, because the calculated doses are low enough not to result in a noticeable radiologically induced health effect. While the nonradiological risks are the greater risks, the results of this study demonstrate that those risks would still not be significant and would be less than the common (nonradiological) environmental risks reported in Table S-4. The results for spent ATF with increased enrichment and higher burnup levels are consistent with the environmental impacts associated with the transportation of fuel and radioactive wastes to and from current-generation reactors presented in Table S-4 of 10 CFR 51.52 (TN250).

Because of the conservative approaches and data used in this study, the NRC staff does not expect the actual environmental effects from the deployment and use of ATF to exceed those calculated in this study for several reasons. A major contributor to the level of the impacts is the number of radioactive material shipments. Longer times between refueling operations will lower the annual number of shipments needed to support an upcoming refueling operation for providing unirradiated ATF assemblies and removing spent ATF assemblies. Additionally, the number of shipments is also tied to the core reload size and the transport package capacity. This last item, transport package capacity, is more of a driver for the number of spent ATF shipments, which would affect the cumulative dose to the exposed population. This study selected the transport packages with the lowest spent fuel assembly capacities to maximize the number of annual spent ATF shipments for conservatism in the evaluation. As the rail sensitivity cases clearly demonstrate, the availability of transport packages that can hold multiple spent ATF assemblies can result in a notable reduction in environmental impacts from normal conditions of transportation. Another cited truck transport package with a greater capacity is the GA-4 package, which is designed to transport up to four intact PWR spent fuel assemblies as authorized contents (NRC 2009-TN8291). In both of these cases, the use of such transport packages would have the effect of reducing the cumulative dose to the exposed population to values below the 3 person-rem of Table S-4.

This study incorporated multiple conservatisms when evaluating accident risks. First, the vehicle accident rates applied in the study are based on accident rates for regular commercial freight shipments. Nuclear fuel shipments are regulated to a stricter standard with inspections, training, and administrative controls due to the potential hazards of the nuclear fuel. This is especially true for SNF shipments with their additional processes to ensure safety and security, such as notifications to the proper authorities along a route, possible security escorts, and monitoring during a shipment. As a result, there has never been a release of radioactive material to the environment that occurred in the U.S. due to transportation of spent fuel. Another factor expected to lower the risks of accidents is the nature of the transport packaging itself. The transport packages considered in this study are referred to as directly loaded fuel packages and were the basis for the release fractions developed by Sprung et al. (2000-TN222). However, several of the dry storage systems and related transport packages developed for SNF involve the placement of the SNF assemblies into an inner canister that is sealed by welding, which would be inserted into the transport package for shipment. As noted in NUREG-2125, this system of SNF packaging is robust enough that there would be no release of radioactive material even under accident conditions (NRC 2014-TN3231). Thus, under accident conditions, members of the public would only have the potential for the physical nonradiological risks from such a transport package system.

Therefore, based on the low risks and conservative nature of this transportation evaluation, the NRC staff has determined that Table S-4 would still bound the environmental impacts from normal conditions and accidents for the transportation of LLRW, unirradiated ATF, and spent ATF for up to 8 wt% U-235 and burnup levels up to 80 GWd/MTU.

For this analysis to be applied in a future licensing action, such as for a license amendment for the deployment and use of ATF with increased enrichment and higher burnup levels, the application would need to confirm the licensee's shipments are bounded by the key parameters of the transport package analyzed here. These parameters are the radionuclide inventory (see Appendix A) based on the applied enrichment and burnup levels, the number of unirradiated fuel shipments, and the number of spent fuel shipments (see Table 3-2). In that case, Table S-4 would apply. If the values associated with the contemplated shipments exceed the above-discussed values, a full description and detailed analysis of the environmental effects of transportation of fuel and wastes as required by 10 CFR 51.52(b) could be performed following the methodology of this NUREG and provided in the application.

4 DECOMMISSIONING

NRC power reactor licenses and the NRC regulations, particularly, 10 CFR 50.82 (TN249), prohibit reactor licensees from abandoning a facility or site. Rather, after cessation of operations, licensees must decommission[1] the facility or site. Decommissioning activities do not include the removal of spent fuel, which is considered to be an operational activity, the storage of spent fuel, or the removal and disposal of nonradioactive structures and materials beyond those necessary to terminate the NRC license (NRC 1996-TN288). Removal of SNF from the spent fuel pool to an ISFSI is overseen by the decommissioning oversight program. With regard to specifically licensed ISFSIs, changes to the ISFSI during decommissioning would be addressed through license amendments. Therefore, the deployment and use of ATF with increased enrichment and higher burnup levels would result in spent ATF being present at a NPP site at the time of decommissioning. The purpose of this section is to address the incremental impacts of deployment and use of ATF, including increased enrichment and higher burnup, and assess the potential change on the impacts of decommissioning as part of the environmental review for a LAR related to the deployment and use of ATF.

The regulations governing decommissioning of power reactors are found in 10 CFR 50.75 (TN249), 10 CFR 50.82 (TN249), and 10 CFR 52.110 (TN251). Under these regulations, decommissioning facilities and sites must meet the radiological criteria for termination of the NRC license in Subpart E of 10 CFR Part 20 (TN283), “Radiological Criteria for License Termination.” Guidance to licensees for the decommissioning of NPPs is provided in RG 1.184 (NRC 2013-TN5470), “Decommissioning of Nuclear Power Reactors” and RG 1.185 (NRC 2013-TN5469), “Standard Format and Content for Post-Shutdown Decommissioning Activities Report.” NUREG-1757 provides the NRC staff’s consolidated decommissioning guidance. As noted in Volume 1 of NUREG-1757, the NRC staff applies NUREG-1748, “Environmental Review Guidance for Licensing Actions Associated with NMSS Programs” (NRC 2013-TN5469) to satisfy NEPA obligations for decommissioning sites where the licensee proposes to release the site for unrestricted use.

In NUREG-0586, the Decommissioning GEIS, the NRC staff evaluated the environmental impacts of nuclear power reactors decommissioning where residual radioactivity at the site is reduced to levels that allow for termination of the NRC license (NRC 2002-TN7254). NUREG-1496, Volume 1, “Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities” (NRC 1997-TN5455) documents results and conclusions related to achieving the objectives of decommissioning. These goals include attaining dose as low as is reasonably achievable (ALARA); reducing dose to preexisting background; meeting the radiological criterion for unrestricted use; performing decommissioning ALARA analysis for soils and structures containing contamination; restricting use and performing alternative analysis for special site-specific situations; and achieving groundwater cleanup (NRC 1997-TN5455). Additionally, the NRC staff evaluated in Section 4.12.2.1 of the 2013 License Renewal GEIS (NRC 2013-TN2654) the environmental impacts only attributable to license renewal for an additional 20 years of operations on the impacts discussed in the Decommissioning GEIS.

[1] Decommissioning is the safe removal of a nuclear facility from service and the reduction of residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license or release of the property under restricted conditions and termination of the license (10 CFR Part 50-TN249).

4.1 Decommissioning Process

The regulations for termination of the license in 10 CFR 50.82(a)(4)(i) (TN249) and 10 CFR 52.110(d)(1) (TN251) require a licensee to submit a post-shutdown decommissioning activity report (PSDAR)[2] to the NRC and copies to the affected State(s) no later than 2 years after permanent cessation of operations. The PSDAR must contain a description of the planned decommissioning activities along with a schedule of their accomplishment; a discussion that provides the reasons for concluding that the environmental impacts associated with site-specific decommissioning activities will be bounded by appropriate previously issued EISs; and a site-specific Decommissioning Cost Estimate, including the projected cost of managing irradiated fuel (10 CFR 50.82(a)(4)(i) (TN249) or 10 CFR 52.110(d)(1)) (TN251).

In meeting those requirements, the licensee would document in its PSDAR the results of the licensee's evaluation of the environmental impacts associated with project-specific decommissioning activities. The evaluation would include a comparison of the site-specific environmental impacts of the proposed decommissioning with the impacts identified in previously issued environmental statements, that is the Decommissioning GEIS (NRC 2002-TN665), NUREG-1496, Volume 1, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities" (NRC 1997-TN5455), and any previous project-specific environmental NEPA licensing documents.

The NRC will review a licensee's PSDAR to determine whether the document contains the information required by 10 CFR 50.82(a)(4)(i) (TN249) or 10 CFR 52.110(d)(1) (TN251), as appropriate. The NRC will also notice receipt of the PSDAR and make it available for public comment in accordance with 10 CFR 50.82(a)(4)(ii) (TN249) or 10 CFR 52.110(d)(2) (TN251), as appropriate. The NRC does not approve the PSDAR because it does not involve a licensing action and because 10 CFR 50.82 (TN249) and 10 CFR 52.110 (TN251) do not require the NRC to approve it. However, if the NRC determines that the information provided by the licensee in the PSDAR does not comply with the regulatory requirements, it will inform the licensee in writing of the additional information required by the regulations and request a response. Additionally, per 10 CFR 50.82(a)(4)(i), if through the review of the PSDAR, the NRC determines that the licensee's proposed activities will result in significant environmental impacts not previously reviewed, in accordance with 10 CFR 50.82(a)(6)(ii), the licensee must change its decommissioning plans or ask for an amendment to authorize those activities before conducting them. NRC review of such an amendment will also include an associated environmental evaluation. As stated in 10 CFR 50.82(a)(8)(ii) (TN249) or 10 CFR 52.110(h)(2) (TN251), as appropriate, licensees are limited in the amount of funds that can be withdrawn from the decommissioning trust fund. The licensee is required to provide updates to the NRC if there are any significant changes to the PSDAR (10 CFR 50.82(a)(7) [TN249] or 10 CFR 52.110(g) [TN251]).

The licensee is required to submit a License Termination Plan amendment application with its final status survey strategy to the NRC at least 3 years before it intends to terminate the license (10 CFR 50.82(a)(9)-(10) [TN249] or 52.110(i)-(j) [TN251]). Before the completion of decommissioning, the licensee conducts a final status survey to demonstrate compliance with criteria established in the approved License Termination Plan and relevant regulatory

[2] The PSDAR is the decommissioning strategy for the NPP. 10 CFR 50.82.(a)(4)(i) (TN249) or 10 CFR 52.110(d)(1) (TN251), as appropriate, specifies what the PSDAR must contain.

requirements (10 CFR 50.82(a)(11) [TN249] or 10 CFR 52.110(k) [TN251]). The NRC staff verifies the survey by one or more of the following: (1) a quality assurance/quality control review, (2) side-by-side or split sampling of a radiological survey of selected areas, or (3) independent confirmatory surveys (NRC 2021-TN8680). When the NRC confirms that the criteria in the License Termination Plan and all other NRC regulatory requirements have been met, the NRC terminates the license, depending on the licensee's decision to use the licensed area (NRC 2013-TN2654). At the end of the decommissioning process (i.e., upon the NRC letter of termination), the site of a nuclear power plant and any remaining structures on the site can be released for unrestricted or restricted use (NRC 2022-TN8031, NRC 2021-TN8680).

4.2 Environmental Impacts from Decommissioning with Accident Tolerant Fuel

Since the deployment and use of ATF would affect the radiological profile of an NPP site, it could result in different decommissioning impacts than previously assessed by the NRC staff, as referenced above, and is assessed here. Cessation of NPP operations would result in the cessation of actions necessary to maintain the reactor, as well as a significant reduction in the workforce. For multiunit sites, with one unit permanently ceasing operations, the NRC staff presumes that the end of that NPP's operations would not immediately lead to the dismantlement of the reactor or other infrastructure, much of which would still be in use to support other units onsite that continued to operate. Further, sites can transition from SAFSTOR to DECON as much as the licensee desires. Under 10 CFR 50.82(a)(3) (TN249) and 10 CFR 52.110(c) (TN251), however, the licensee must decommission the site within 60 years of permanent cessation of operations, unless the Commission approves an extension beyond 60 years. For LWRs, it takes approximately 8 to 10 years in DECON to completely decommission a site for license termination. Even for sites with just one unit, some facilities would remain in operation to ensure that the site would be maintained in safe shutdown condition or for other reasons. For example, electrical generators might continue to operate as synchronous condensers to stabilize voltage on the bulk electricity grid to which the reactor was connected. Deployment and use of ATF would not affect these activities.

Three decommissioning options were analyzed in the Decommissioning GEIS (NRC 2002-TN7254) and are referenced in this section: DECON (immediate decontamination), SAFSTOR (SAFe STORage – deferred dismantling), or ENTOMB (entombment – permanent encasement of radioactive contaminants). In the DECON option, the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed and safely buried in a LLRW landfill or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations. In the SAFSTOR option, the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated to levels that permit release for restricted or unrestricted use. Finally, with the ENTOMB option, radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombment structure is appropriately maintained, and continued surveillance is sustained until the radioactivity decays to a level permitting unrestricted release of the property. However, the ENTOMB option is not preferred and has not been implemented by an NRC licensee.

In the Decommissioning GEIS, the NRC staff assessed the following environmental issues for their environmental impacts during decommissioning:

- land use
- visual resources
- air quality

- noise
- geology and soils
- water resources—surface water and groundwater
- ecological resources
- historic and cultural resources
- socioeconomics
- human health
- environmental justice
- waste management and pollution prevention

Since the deployment and use of ATF with increased enrichment and higher burnup levels would not result in major changes to the NPP itself, such as the physical structure, footprint, or supporting plant operational and auxiliary systems, there would not be any additional decommissioning activities as a result of deployment and use of ATF for most environmental issues. Thus, many of the decommissioning impacts discussed in Section 4, "Environmental Impacts of Decommissioning Permanently Shutdown Nuclear Power Reactors," of the Decommissioning GEIS (NRC 2002-TN665) for the above environmental issues remain the same or are specific to a site (e.g., cultural resources) for the deployment and use of ATF irrespective of the enrichment and burnup levels of the spent ATF. Therefore, impact assessments discussed in the Decommissioning GEIS are expected to remain unchanged for land use, visual resources, air quality, noise, geology and soils, water resources, ecological resources, historic and cultural resources, socioeconomics, and environmental justice; the impact assessments for these topics are incorporated here by reference. The remainder of the section addresses the decommissioning impacts from the deployment and use of ATF for the remaining two environmental issues—human health along with waste management and pollution prevention.

4.2.1 Human Health

With the termination of plant operations, there would be a period of time between when a reactor stops operation and the implementation of the active decommissioning of the plant, which could range from months to years. During that period, the reactor would be placed in a cold shutdown condition and maintained. The spent fuel would be removed from the core and put in the spent fuel storage pool and later transferred to dry cask storage in an ISFSI. Also, during this time, workers would continue to receive radiation exposure during work activities related to placing the reactor in shutdown status. Because of the longer times between refueling operations as a result of increased enrichment and higher burnup levels, there would be a lower number of fuel assemblies to manage compared to existing LWR fuels. Hence, the licensee would process fewer spent fuel assemblies on an annual basis resulting in lower accumulated occupational radiation doses. Therefore, the NRC staff concludes the accumulated occupational exposures during decommissioning would be lower with ATF with increased enrichment and higher burnup, and the analysis in the Decommissioning GEIS would still be bounding for the ATF technologies, including increased enrichment and higher burnup.

Even though the NPP would have ceased operation, there would be some residual radioactive gaseous and liquid effluent releases into the environment that could result in some radiation exposure to the public. This exposure would continue during decommissioning because radioactive materials other than SNF are processed for disposal and storage. The regulatory requirements and dose limits during this period for workers and the public are the same as those for operating reactors (see Section 3.9.1.1 of the 2013 License Renewal GEIS, NRC

2013-TN2654). With regard to occupational exposure, spent ATF, including ATF with increased enrichments of up to 10 wt% U-235 and higher burnup levels up to 80 GWd/MTU, must be stored within the ISFSI under the same 10 CFR Part 20 and Part 72 regulations for radiological protection as for current SNF. At the time of decommissioning, the licensee can manage the process of transferring spent ATF from a likely full spent fuel pool to an ISFSI in ways similar to those for current SNF (e.g., longer time in the spent fuel pool to allow for lower decay heat levels at the time of transfer) to ensure regulatory requirements, such as ALARA, are met. The radiological impacts on workers and members of the public during the period of decommissioning are expected to be equal to or less than the exposure to radiation during the operation of the NPP with the impacts decreasing over time as systems, structures, and components are decontaminated, dismantled, appropriately packaged, and shipped to a radiological disposal site. Because decommissioning facilities would follow the same regulations with the same dose limits even if they used fuels with increased enrichment and higher burnup levels, these radiological impacts on workers and members of the public would occur irrespective of whether the nuclear fuel was conventional LWR fuel or ATF and, therefore, the analysis in the Decommissioning GEIS would still be bounding for the ATF technologies.

The deployment and use of ATF has no effect on nonradiological impacts because the deployment and use of ATF does not change the chemical control and operation of other plant systems. Therefore, the public's exposure to chemical and microbiological hazards associated with decommissioning operations, such as from the cooling system, would not be different from those of decommissioning activities before ATF deployment and use. For example, as discussed in the Decommissioning GEIS, the cessation or reduction of cooling system operations with reduced thermal discharges over time results in lower public health risks from microbiological hazards compared to the operating period. As another example, as discussed in the Decommissioning GEIS, the plant workers might be exposed to chemical, microbiological, and other hazards during decommissioning, but the hazards would be controlled for all plants and bounded by the hazards during operations. Therefore, the nonradiological impact analysis in the Decommissioning GEIS would bound the ATF technologies.

In conclusion, because the termination of operations at plants that deployed and used ATF technologies with increased enrichments up to 10 wt% U-235 and higher burnup levels up to 80 GWd/MTU would not result in any significant physical changes during decommissioning and there would be less or the same radiological exposure, the impacts from decommissioning on human health would be less than or the same as those considered in the Decommissioning GEIS and, therefore, they would be bounded by the Decommissioning GEIS.

4.2.2 Waste Management and Pollution Prevention

During decommissioning activities, additional waste might accumulate at the site or the radioactivity of some components undergoing decommissioning might be slightly higher at the end of the operating period due to refurbishment activities. The amounts of certain types of waste (e.g., LLRW) generated from decommissioning due to the deployment and use of ATF could be more than the amounts generated with the use of conventional fuels.

There might be small differences in the quantities and characteristics of the waste that would be generated during decommissioning from the deployment and use of ATF technologies. The level of radioactivity from neutron activation for materials in and around the core would depend on the timing of decommissioning activities (Krall et al. 2022-TN8682). The deployment and use of ATF could result in higher levels of radioactivity as a result of greater amounts of radionuclides due to higher burnup levels. This could affect the quantity of Class A, B, and C

LLRW due to the potentially greater radionuclide inventory in the fuel assemblies. However, it would likely have little effect on the amount of greater-than-Class C LLRW at the site since that waste is mainly a result of neutron activation (PNNL 1984-TN8683). Assuming that the ATF SNF would continue to be stored onsite, there would also be less spent fuel to manage due to the longer periods of time between refueling operations (e.g., extension of operations from 18 months to 2 years). This change would primarily be observable as reduced loading in an ISFSI prior to defueling the reactor to the spent fuel pool and during the ultimate transfer of all assemblies to the ISFSI's dry cask system. Because all radioactive waste must be handled during decommissioning in accordance with NRC regulations no matter the level of enrichment and burnup (and the NRC staff has determined the current regulatory scheme is sufficient to regulate ATF), and the size and structure of ATF assemblies would be similar to or the same as the existing fuel assemblies, the deployment and use of ATF would not significantly alter the practices licensees employ to manage the wastes and the resulting impacts during decommissioning.

The decommissioning activities would be designed and implemented in ways to prevent pollution and minimize the amount of waste generated irrespective of the type of nuclear fuel including ATF (10 CFR Part 20-TN283). The procedures and practices implemented would be aimed at preventing or minimizing gaseous and liquid releases to the environment and the quantities of waste generated. The NRC staff also analyzed the offsite transportation of equipment and wastes from a power plant undergoing decommissioning in the Decommissioning GEIS (NRC 2002-TN7254), and the impact was found to be small. Due to longer refueling times as a result increased enrichment and higher burnup levels, the overall number of spent fuel assemblies at the time of decommissioning would be less than for the existing LWR conditions expected at decommissioning resulting in smaller ISFSI. No significant changes to decommissioning waste management activities are expected from the deployment and use of ATF.

4.3 Greenhouse Gas Emissions

PNNL assessed the contribution decommissioning makes to GHG emissions as part of an assessment for the NRC entitled "Assumptions, Calculations, and Recommendations Related to a Proposed Guidance Update on Greenhouse Gases and Climate Change" (Chapman 2012-TN2644). PNNL assessed two sources of GHG emissions during decommissioning activities, namely decommissioning equipment and decommissioning workforce, over a 10-year period for completing the decommissioning of a 1,000 MWe NPP. For decommissioning equipment, Chapman (2012-TN2644) estimated 19,000 MT CO_2e and 8,400 MT CO_2e for the decommissioning workforce over 10 years. Thus, the annual CO_2e emissions from all decommissioning activities would be approximately 2,740 MT CO_2e e per year, a very small fraction of the 2020 total CO_2e emissions for the United States (EPA 2023-TN8681). Additionally, as discussed in the Decommissioning GEIS, various systems associated with reactors contain gases that are of environmental concern (NRC 2002-TN7254). For example, some gases used in refrigeration systems and fire-suppression systems have been identified as ozone-depleting compounds. The deployment and use of ATF with increased enrichment and higher burnup levels would not alter the use of or the quantity of such ozone-depleting compounds. Venting of these gases to the atmosphere is prohibited by law. Standard methods exist to purge systems containing these gases and limit releases to the environment to insignificant quantities. Other fire-suppression and refrigeration systems may contain GHGs. The quantities of these gases at a nuclear plant are generally small in comparison with the quantities of GHGs released hourly by a fossil-fuel combustion plant used for heating or power generation. The impacts of ozone-depleting gases and GHGs are global rather than local.

Therefore, it is unlikely that releases of ozone-depleting or greenhouse gases during decommissioning of any NPP will be detectable or destabilize the environment, whether ATF technologies are at the site or not.

4.4 Accident Tolerant Fuel Decommissioning Conclusions

The deployment and use of ATF technologies with increased enrichment and higher burnup levels do not result in physical changes to an NPP and could create less spent fuel over time than a facility that uses existing nuclear fuel, while providing the same energy output. Therefore, for most environmental issues evaluated, the decommissioning impacts would be the same as or slightly less than the impacts associated with decommissioning NPPs operating with the existing fuel. Thus, the analysis in the 2013 License Renewal GEIS and Decommissioning GEIS would bound an NPP deploying ATF undergoing decommissioning.

In SRM-SECY-18-0055 (NRC 2021-TN8079), the Commission directed the NRC staff to update the Decommissioning GEIS to reflect current decommissioning practices and lessons learned from previous reviews. Additionally, the NRC staff was also directed to provide specific guidance for environmental issues that cannot be generically resolved in the Decommissioning GEIS. Thus, the NRC staff expects the Decommissioning GEIS and guidance updates could build upon the analysis from this study to specifically address the decommissioning of a LWR deploying and using ATF.

5 CONCLUSION

To support efficient and effective licensing reviews of requests to use ATF and to reduce the need for complex site-specific environmental reviews for each ATF LAR, this study evaluated the reasonably foreseeable impacts of deploying and using near-term, first- and second-generation ATF technology with increased enrichment and higher burnup levels on the uranium fuel cycle, transportation of fuel and waste, and LWR decommissioning (e.g., bounding analysis). The NRC staff determined that the ATF technologies analyzed in this NUREG (i.e., coated cladding, doped pellets, and FeCrAl cladding) would have the same or fewer environmental effects than traditional fuel under conditions of spent fuel storage and transportation. The NRC staff evaluated the impact of increased enrichment and higher burnup levels by assessing and applying NRC-sponsored ATF technology reports, prior environmental reviews, transportation studies, and new or updated data sources to determine the bounding (generic) environmental impacts of deploying ATF technologies with increased enrichment and higher burnup levels in LWRs.

For the uranium fuel cycle, there have been significant changes in the front-end processes and to NRC-licensed facilities since the publication of WASH-1248. The most notable examples are extraction of uranium from the ground using in-situ recovery instead of traditional mining, performing all enrichment using gaseous centrifuges instead of gaseous diffusion, and electricity generation moving significantly away from the use of coal. Thus, the NRC staff concluded that the front-end of the uranium fuel cycle involving ATF technologies with increased enrichment up to 10 wt% U-235 will have environmental effects bounded by the environmental data provided in Table S-3 under 10 CFR 51.51 (TN250).

Regarding the back-end of the uranium fuel cycle, the current practices of long-term management of SNF would still apply to the deployment and use of ATF with higher burnup levels. For example, as with current LWR spent fuel, the cooling time in a spent fuel pool for ATF with higher burnup levels would need to be 1 year (10 CFR 72.2 (a)(1)) (TN4884) and meet the thermal limits of a licensed dry cask storage system prior to transfer to an ISFSI. A benefit of the deployment and use of ATF with the higher burnup levels would be the longer times between refueling operations, which would lessen the average annual rate of spent ATF assemblies being placed into the spent fuel pools and ultimately transferred to an ISFSI. Thus, lengthening the time between refueling operations also lengthens the time before expansion of an ISFSI would be necessary because of the overall reduction of the number of spent fuel assemblies being placed into dry storage over the time of operations. This would reduce the environmental impacts beyond those that would occur with current fuel; the impacts of ATF in this regard would be bounded by prior NRC environmental evaluations.

Regarding the deployment and use of ATF with increased enrichment and higher burnup levels, the NRC staff determined that the analyses in the Continued Storage GEIS were sufficiently conservative to bound the impacts such that any variances that may occur from site to site are unlikely to result in environmental impact determinations that are greater than those presented in the Continued Storage GEIS. Therefore, since spent ATF would conform with the analysis of the Continued Storage GEIS (NRC 2014-TN4117), the Continued Storage GEIS would still be bounding for the environmental impacts of spent ATF.

The NRC staff's re-evaluation of the environmental effects from the transportation of unirradiated ATF and waste demonstrates that the deployment and use of ATF would be bounded by Table S-4 for up to 8 wt% U-235 and up to burnup levels of 80 GWd/MTU, especially if transport packages with higher capacities are used. As previously noted, this re-evaluation is conservative for various reasons. The level of conservatism is demonstrated by the rail shipment sensitivity calculations, which show that the dose risks to members of the public can be significantly reduced by using transport packages that can hold a large number of spent ATF assemblies, thereby reducing the number of shipments. Because of the uncertainty in fuel-cladding gap releases at higher burnup levels above 62 GWd/MTU from the previous study reported by Ramsdell et al. (2001-TN4545), an assessment of available data was performed to bound the expected increased gas gap source term and fission product releases from failed fuel as burnup increases to 72 and 85 GWd/MTU levels. While the release fractions were greater for a number of severity cases than those provided by Sprung et al. (2000-TN222), especially for particulates, the overall risks were still lower than prior studies, such as that of Ramsdell et al. (2001-TN4545), due to items such as changes in the dose calculations in the RADTRAN code to remove previous dose conservatisms.

In the case of decommissioning, the expected impacts from deployment and use of ATF with increased enrichment and higher burnup levels would be the same as or slightly less than the impacts associated with decommissioning NPPs operating with the existing fuel. Therefore, the existing analyses in the 2013 License Renewal GEIS and the Decommissioning GEIS bound the impacts from the deployment and use of ATF. Additionally, the expected Decommissioning GEIS and guidance updates could build upon the analysis from this study to specifically address the decommissioning of a LWR deploying and using ATF. Therefore, based on findings in this study, the NRC staff concludes that the reevaluated findings addressing near-term, first- and second-generation ATF technologies (i.e., coated cladding, doping, and FeCrAl cladding) indicate the environmental effects associated with deploying and using ATF would be bounded by the NRC staff’s prior decommissioning analysis for enrichments up to 10 wt% U-235 and extending assembly averaged burnup to 80 GWd/MTU.

The results of this analysis could serve as a reference in helping to address the environmental impacts in ATF licensing actions without a detailed site-specific transportation analysis, as long as the ATF is within the bounds and assumptions of the analyses within this NUREG (e.g., enrichment and burnup levels with the associated fuel assembly radionuclide inventory). It is important to note that the purpose of this study in future ATF LAR application reviews is to provide an environmental evaluation that could support the environmental review for a specific LAR, for a specific site, and specific reactor parameters for a qualified type of ATF.

In conducting a generic evaluation, the NRC staff based its analysis on certain conditions that may or may not be present at specific sites. To rely on the analysis in this study, applicants and licensees should assess whether the site-specific conditions meet those assumed conditions. In particular, applicants and licensees should discuss:

- the expected refueling cycle times (e.g., 24 months)
- whether the enrichment level for the type of ATF in the LAR application is within those discussed in this document (i.e., 10 wt% U235 for uranium fuel cycle and 8 wt% U-235 for transportation of spent ATF)
- whether the maximum assembly averaged burnup level is no greater than 80 GWd/MTU
- the maximum radionuclide inventories in a spent ATF assembly based on power history, enrichment, and assembly average burnup level in the ATF LAR for whether the quantity of the radionuclides are within those shown in Table A-1 of this NUREG

- whether the number of annual unirradiated and spent ATF shipments over the refueling cycle time being requested in the LAR application based on the expected transport package fall within the number of shipments discussed in this NUREG in Table 3-2
- whether the applicant intends to use a sealed canister for the type of dry cask storage system at the site's existing ISFSI and whether such a canister would also be used in a certified transport package
- whether the transport mode of the expected certified transport package aligns with the modes considered in this NUREG
- whether the expected decommissioning environmental impacts, after deployment and use of ATF, would be bounded by the impacts discussed in Section 4 of this NUREG and as addressed in the Decommissioning GEIS

After verifying the applicability of this NUREG in a specific ATF LAR application, a licensee may incorporate it by reference in the ATF LAR, and the NRC staff may incorporate it by reference in its associated environmental evaluation. If any of these applicability criteria are not met, an applicant may be able to rely on the information in this NUREG in its environmental report, but it would have to demonstrate that the specific LAR would have environmental effects equal to or less than those discussed in this NUREG. Else, if in a future licensing action where the enrichment and burnup levels are greater than the previously mentioned values, an applicant or licensee can apply the methodology and guidance of NUREG-2266 for completing the needed revised analysis for the higher enrichment and burnup levels.

As far as the NRC staff is aware, the maximum enrichment level industry is interested in pursuing for use with any ATF technology is no greater than 10 wt% U-235. If seeking approval for use of fuel with enrichment levels exceeding 10 wt% U-235 enrichment, an applicant would need to assess uranium fuel cycle and decommissioning environmental impacts in its ATF LAR. This assessment could apply the rationale in Sections 2 and 4, respectively. Industry may be interested in exceeding the assembly average 80 GWd/MTU burnup levels. In that case, the methodologies in Sections 2 (uranium fuel cycle) and 4 (decommissioning) can be applied for levels exceeding 80 GWd/MTU. As the transportation analysis in this NUREG does not apply beyond enrichments of 8 wt% U-235 or burnup levels beyond 80 GWd/MTU, a new transportation analysis would be necessary in any ATF LAR seeking approval above 8 wt% U-235 and 80 GWd/MTU. That said, the methodology in Section 3 can be applied using the data sources documented in Appendix A through Appendix D.

6 REFERENCES

10 CFR Part 20. *Code of Federal Regulations*, Title 10, *Energy*, Part 20, “Standards for Protection Against Radiation.” TN283.

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7 LIST OF PREPARERS

This NUREG on “Environmental Effects of Accident Tolerant Fuel with Increased Enrichment and Higher Burnup Levels” was prepared by staff at the NRC and at PNNL (Table 7-1).

Table 7-1 List of Preparers

Name	Education/Expertise	Contribution
Donald E. Palmrose	B.S., Nuclear Engineering; M.S., Nuclear Engineering; Ph.D., Nuclear Engineering; 36 years of experience in project management, operations, research, and technical review expertise in NRC licensing reviews, NEPA assessments and documentation, regulatory analysis, risk assessments, nuclear safety analysis, and radiation protection	Lead Project Manager and Author
Seshagiri Rao Tammara	B.S., Chemical Engineering M.S., Chemical Engineering M.S., Environmental Engineering; 49 years of experience in chemical and environmental engineering	Author
Kenneth J. Geelhood	B.S., Mechanical Engineering; M.S., Materials Science; 21 years of experience in nuclear fuel thermal-mechanical performance	Author (Appendix B)

(a) PNNL is managed for the DOE by Battelle Memorial Institute.

APPENDIX A

SPENT ACCIDENT TOLERANT FUEL RADIONUCLIDE INVENTORIES

The transportation package radionuclide inventories applied in this study were derived from an existing Office of Nuclear Regulatory Research project to assess the effects of accident tolerant fuel (ATF) with increased enrichments and high burnup. As discussed in Section 1.4, traditional fuel at high burnups and high enrichments are expected to be bounding for near-term ATF technologies. Although future ATF and traditional assembly designs may have slightly different dimensions, these calculations are expected to be generally applicable where these differences are not expected to significantly alter these findings. The assessments under this project were performed by staff at Oak Ridge National Laboratory (ORNL) for selected representative light-water reactor (LWR) fuel designs. The project has multiple phases. Phase 1 focuses on the lattice physics parameters and used fuel nuclide inventory changes for typical pressurized water reactor (PWR) and boiling water reactor (BWR) designs (i.e., a conventional Westinghouse 17 × 17 PWR design) (Hall et al. 2021-TN8084) and for a conventional GE14 10 × 10 BWR design with GNF-2 part length rod patterns to model a modern BWR assembly design (Cumberland et al. 2021-TN8085).

The primary Phase 1 investigation tool is SCALE, specifically the Polaris sequence. Polaris is SCALE's 2-dimensional lattice physics tool for LWR analysis, and the Phase 1 work uses Evaluated Nuclear Data File (ENDF)/B-VII.1 cross sections (Wieselquist et al. 2020-TN8090). In addition, Phase 1 performed front-end analysis of uranium hexafluoride (UF_6) transportation packages using SCALE's Criticality Safety Analysis Sequence (Hall et al. 2020). Phase 2 continued with additional studies to identify the effects of loading LEU+ fuel (i.e., moderate increases beyond 5 weight percent [wt%] of uranium-235 [U-235] enrichment) with increased burnup on the thermal and shielding performance of current dry storage cask systems (Kucinski et al. 2022-TN8091). Phase 2 calculations were performed using the U.S. Nuclear Regulatory Commission (NRC) core simulator PARCS and SCALE/Polaris, ORIGAMI, and MAVRIC codes. Source terms, shielding, and peak cladding temperature calculations were performed using contemporary cask designs from the Used Nuclear Fuel – Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) tool (Lefebvre et al. 2017-TN8092).

The fuel assembly radionuclide inventory data, after 5 years of cooling for at least 147 radionuclides, were generated for set enrichments and assembly averaged burnup levels. For Phase 1, radionuclide inventory data were generated for spent nuclear fuel (SNF) with enrichments of 5 and 8 wt% U-235 and assembly averaged burnup levels of 60 and 80 GWd/MTU along with various numbers of integral fuel burnable absorber rods. Radionuclide inventory data from the Phase 2 assessment were generated for SNF with enrichments from 4.2 up to 7.9 wt% U-235 and assembly averaged burnup levels of 52 and 72 GWd/MTU (see Appendix A of Kucinski et al. [TN8091] for additional details).

To perform a bounding and conservative accident analysis of the transportation of spent ATF, the NRC staff assessed the provided radionuclide inventory data generated by ORNL to select the maximum curie content for the radionuclides of concern. First, using the approximately 39 radionuclides applied in past new reactor environmental transportation evaluations, the NRC staff selected the maximum curie value for each of these radionuclides from the Phase 1 and 2 data. Of note from assessing these data is the variation between enrichment and burnup levels where some radionuclides had a maximum curie value at a lower enrichment and burnup level rather than that found for the highest enrichment and burnup levels. Regardless of the enrichment and burnup level, the maximum radionuclide curie value was selected for BWR and PWR fuel assemblies and normalized to 0.5 MTU to be consistent with the truck transportation analysis of WASH-1238 (AEC 1972-TN22).

While NRC-Radioactive Material Transport (RADTRAN) has a data library for approximately 150 radionuclides, the NRC staff limited the number of radionuclides necessary for the NRC-RADTRAN calculations to those that have a significant contribution to the radiological doses. By using a radionuclide's A2 value as an indicator of the health effect of that radionuclide, the NRC staff determined that 11 radionuclides were significant contributors to radiological dose. These radionuclides were verified in NRC-RADTRAN runs where radionuclides with lower curie inventories were incrementally removed, results were compared showing no change, and this process was continued until there was a change in results that yielded the remaining 11 radionuclides with the largest A2 values. The krypton-85 (Kr-85, a gas) and a crud component (i.e., cobalt-60 [Co-60]) were also included since occurrence of their release is expected (i.e., Kr-85) or it is already on the outside of a fuel assembly (i.e., Co-60 in crud). Table A-1 presents the resulting list of radionuclides and their bounding inventory in curies on a per 0.5 MTU fuel assembly basis to be applied in the NRC-RADTRAN calculations that contribute to 99.99 percent of the radiological doses.

Table A-1 Radionuclide Inventory Selected for NRC-RADTRAN Accident Tolerant Fuel Calculations

A2 + Radionuclides	Chemical Group[(a)]	Bounding 0.5 MTU Inventory (Curies)	Radionuclide Inventory Source
Co-60	Crud	4.38E+03	Ramsdell et al. (2001-TN4545)
Kr-85	Gas	8.04E+03	Hall et al. 2021-TN8084
Sr-90	Particle (Part)	8.07E+04	Hall et al. 2021-TN8084
Y-90	Part	8.07E+04	Hall et al. 2021-TN8084
Ru-106	Ru	1.76E+04	Wieselquist et al. 2020-TN8090
Cs-134	Cs	5.05E+04	Hall et al. 2021-TN8084
Cs-137	Cs	1.10E+05	Hall et al. 2021-TN8084
Pu-238	Part	7.98E+03	Hall et al. 2021-TN8084
Pu-239	Part	2.61E+02	Hall et al. 2021-TN8084
Pu-240	Part	3.99E+02	Wieselquist et al. 2020-TN8090
Am-241	Part	1.12E+03	Hall et al. 2021-TN8084
Pu-241	Part	1.03E+05	Hall et al. 2021-TN8084
Cm-244	Part	1.42E+04	Hall et al. 2021-TN8084

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^2 is indicated by 5.02E+02.
Am = americium; Ci = curies; Cm = curium; Co = cobalt; Cs = cesium; Kr = krypton; Sr= strontium; Ru = ruthenium; Pu = plutonium; Y = yttrium.
(a) Chemical groups applied in NRC-RADTRAN is based on Chapter 7 and Table 7.31 of Sprung et al. (2000-TN222).

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APPENDIX B

EXAMINATION OF RADIOLOGICAL RELEASE FRACTIONS DUE TO HIGHER BURNUP LEVELS

B.1 Summary of Changes

As discussed in Section 1.4, traditional fuel at high burnup and high enrichment is expected to be bounding for near-term accident tolerant fuel ATF technologies. For example, as noted in the study by Hall et al. (2021-TN8286), calculations of isotopic content changes associated with chromium (Cr)-coated cladding and doped pellets only demonstrated minor effects of ATF vs non-ATF for enrichments of 5 and 10 wt% U-235 and burnup of 62 and 80 GWd/MTU. In the case of FeCrAl cladding, as discussed in Section 1.4, additional performance data would be provided to clarify the effect of this cladding regarding release fractions if this ATF technology is going to be deployed. Although future ATF and traditional assembly designs may have slightly different dimensions, these calculations contain some conservatism and are expected to be generally applicable where these differences are not expected to significantly alter these findings.

Increasing nuclear fuel burnup will affect the dose consequences of spent fuel package accidents in a number of different ways. The most obvious is the increase in fission products in the fuel due to the occurrence of greater number of fissions. In addition, as burnup progresses, many of the fission products and primarily the gaseous fission products diffuse out of the pellet into the fuel-cladding gap and upper plenum, thereby leading to an increase in the total moles of noble gas (helium [He], xenon [Xe], and krypton [Kr]) that would be released into the package in case of fuel-cladding failure.

In addition to these two mechanisms, three other mechanisms are affected by burnup that could affect the dose consequences, as listed below:

1. Cladding embrittlement. During the time spent in-reactor, the fuel-cladding is exposed to high-temperature (550–600 degrees Fahrenheit [°F], 288–316 degrees Celsius [°C]) water and a slow reaction between the cladding metal and the water results in the formation of a zirconium oxide layer that somewhat reduces the cladding wall thickness. Additionally, a significant fraction of the hydrogen generated by this reaction is absorbed into the cladding, forming a brittle zirconium hydride phase. This zirconium hydride phase, as well as the effect of fast neutron damage, which increases with burnup, leads to a marked decrease in the cladding strain capability.
2. Fuel fragmentation. Above a local burnup of about 50 GWd/MTU, the fuel pellet exhibits a high burnup rim structure characterized by sub-micron grains and high gas porosity. The thickness of this rim structure increases with burnup such that about 10 percent of the volume of the pellet consists of this rim structure at a rod-average burnup of 72 GWd/MTU. This structure is vulnerable to fragmentation into small particles in the event of a severe mechanical or thermal event and will result in an increase in the fuel particulate releases for some accidents.
3. Additional diffusional release of fission products. Some of the postulated accidents include the fuel being subjected to high temperature for a significant period of time, and during this time, it is possible that additional fission products (cesium [Cs], ruthenium [Ru], Xe, and Kr)

may diffuse out of the fuel pellets. Additionally, failed fuel may become available for release into the cask interior.

In 2000, Sandia National Laboratories performed a study to examine the radiological risk of spent fuel with burnup up to 60 GWd/MTU (NUREG/CR-6672, Volumes 1 and 2, Sprung et al. 2000-TN222) using the Radioactive Material Transport (RADTRAN code). The initial approach to estimating the risk of spent fuel with burnup up to 72 GWd/MTU is described below. The results of this approach would be considered an initial estimate that is subject to the following limitations:

- This approach follows the methodology of NUREG/CR-6672 and only alters input parameters because they would change with higher burnup. It does not examine the validity of this method to adequately predict dose consequences of fuel up to 60 GWd/MTU.
- Given a lack of data, the change in some parameters is uncertain and, in these cases, PNNL recommends limiting values, which would result in a conservative estimate of dose.

The changes to the RADTRAN input are two-fold. First, there are changes to the accident source term (radionuclide inventory and fuel-cladding gap inventory) that would be applied to every event. Second, there are changes that are event specific (based on impact velocity and fuel temperature) and therefore the changes for increased burnup will be different for different events. Table B-1 shows the overall approach that is used for each item that is expected to change and there is a corresponding section in this study where the determination of appropriate parameters is provided.

Table B-1 Items Addressed in High Burnup Spent Fuel Analysis

Source Term	**Event Specific** ***Mechanical Only***	**Event Specific:** ***Mechanical and Temperature***	**Event Specific:** ***Temperature Only***
Radionuclide inventory (calculated with ORIGEN code)	Fraction fuel rods failed	Fraction fuel rods failed	Fraction fuel rods failed
Fuel-Cladding Gap Inventory (calculated with FAST code)	Particulates Xenon+Krypton	Particulates Xenon+Krypton Cesium/Rubidium	Particulates Xenon+Krypton Cesium/Rubidium

B.2 Cases

NUREG/CR-6672 examined 18 truck events (cases) for pressurized water reactor (PWR) and boiling water reactor (BWR) fuel rods that resulted in a release. The two parameters that have an impact on the fuel performance are impact velocity and temperature. Table B-2 shows the ranges of these parameters for each event based on Section 7.2.6 and Table 7-10 of NUREG/CR-6672 (Sprung et al. 2000-TN222).

B.3 Radionuclide Inventory

The Oak Ridge National Laboratory has produced a report (ORNL/TM-2022/1841, Kucinski et al. 2022-TN8091) that calculates the radionuclide inventory for a PWR fuel assembly with a rod-average burnup of 72 GWd/MTU. See Appendix A of this document.

Table B-2 Pressurized Water Reactor Accident Cases

Case	Temperature (°C)	Impact Velocity (mph)
1	20	>120
2	20–350	30–60
3	350–750	30–60
4	750–1000	30–60
5	20–350	60–90
6	350–750	60–90
7	750–1000	60–90
8	20–350	90–120
9	350–750	90–120
10	750–1000	90–120
11	20–350	>120
12	350–750	>120
13	750–1000	>120
14	750–1000	30–60
15	750–1000	60–90
16	750–1000	90–120
17	750–1000	>120
18	750–1000	0

°C = degrees Celsius; mph = miles per hour.

B.4 Fuel-Cladding Gap Inventory

The existing analysis assumes that all the rods in the casks contain four times as much gas as the gas in the cask. It also assumes that the cask is pressurized to 1 atmosphere (atm), so if all the rods fail, the pressure in the casks will increase by 4 atm to a value of 5 atm. The following two sections describe fuel-cladding gap inventory by reactor type.

B.4.1 Pressurized Water Reactor

The cask assumed for this analysis contains 6 moles (mol) of gas (0.147134 m^3, 1 atm, 300 Kelvin [K]). FAST[1] calculates that a PWR rod irradiated to 72 GWd/MTU contains between 0.025 and 0.054 mol of gas depending on the fission gas release fraction (unirradiated rods contain 0.019 mol of gas). Note that FAST uses the same fission gas release model for doped and undoped fuel and shows a negligible impact of cladding type on the fission gas release. There are 264 rods in each 17 × 17 fuel assembly, and the cask contains one assembly. Therefore, if all the rods contain the maximum amount of gas and were to rupture, they would release 14 mol of gas which is not greater than four times the moles of gas in the cask.

Because of this analysis, we recommend retaining 1 to 4 atm for the pressure increase from ruptured fuel rods.

[1] A computer code for the calculation of steady-state and transient, thermal-mechanical behavior of oxide fuel rods for high burnup.

The input to RADTRAN uses four expansion factors that are a function of the pressure differential discussed above and the rod failure fractions that are discussed in an upcoming section.

Table B-3 shows the updates that would be used for the HBU analysis for a PWR.

Table B-3 Updates to Expansion Factors for Pressurized Water Reactors

Parameter	Impact Velocity (mph)	Original Value	New Value
F1	>90	0.184	0.184
F1	60–90	0.274	0.184
F1	30–60	0.460	0.307
F2	All	0.609	0.609
F3	>90	0.112	0.112
F3	60–90	0.167	0.112
F3	30–60	0.280	0.187
F4	>90	0.804	0.804
F4	60–90	0.304	0.804
F4	30–60	0.201	0.268
F5	>90	0.200	0.200
F5	60–90	0.298	0.200
F5	30–60	0.500	0.333

mph = miles per hour.

B.4.2 Boiling Water Reactor

The cask assumed for this analysis contains 6 mol of gas (0.147134 m^3, 1 atm, 300 K). FAST[2] calculates that a BWR rod irradiated to 72 GWd/MTU contains about 0.079 mol of gas depending on the fission gas release fraction (unirradiated rods contain 0.071 mol of gas). There are 92 fuel rods in each 10 × 10 fuel assembly, and the cask contains two assemblies. Therefore, if all the rods contain the maximum amount of gas and were to rupture, they would release 15 mol of gas, which is not greater than four times the moles of gas in the cask.

Because of this analysis, we recommend retaining 1–4 atm for the pressure increase from ruptured fuel rods.

The input to RADTRAN uses four expansion factors that are a function of the pressure differential discussed above and the rod failure fractions that are discussed in an upcoming section.

Table B-4 shows the updates that would be used for the high burnup analysis.

[2] A computer code for the calculation of steady-state and transient, thermal-mechanical behavior of oxide fuel rods for high burnup.

Table B-4 Updates to Expansion Factors for Boiling Water Reactors

Parameter	Impact Velocity (mph)	Original Value	New Value
F1	>90	0.184	0.184
F1	60–90	0.511	0.354
F1	30–60	0.821	0.742
F2	All	0.609	0.609
F3	>90	0.112	0.112
F3	60–90	0.311	0.215
F3	30–60	0.500	0.452
F4	>90	0.804	0.804
F4	60–90	0.191	0.236
F4	30–60	0.165	0.169
F5	>90	0.200	0.200
F5	60–90	0.556	0.385
F5	30–60	0.893	0.806

mph = miles per hour.

B.5 Fuel Rod Failure Fraction

The existing analysis performs a finite element analysis calculation for various drop events and determines the maximum strain experienced in each fuel rod. Each fuel rod is assigned a strain limit and is assumed to fail if the predicted strain exceeds this value. The existing analysis assumed a decreasing failure strain limit with burnup that was 1 percent for rods with 55–60 GWd/MTU burnup. However, the cask analyzed was not filled with rods at this burnup level, but a distribution of burnup with the lower burnup rods having a greater strain to failure.

Modern fuel rods that will be irradiated to 72 GWd/MTU will have more advanced zirconium-alloy cladding than the historic Zircaloy-4 that was used for PWR fuel. The M5, optimized ZIRLO, and AXIOM all exhibit superior corrosion and hydrogen pickup relative to Zircaloy-4, such that a 1 percent failure strain limit for these rods at 72 GWd/MTU is reasonable for PWRs. Modern Zircaloy-2 variants with controlled chemistry and ZIRON all exhibit superior corrosion and hydrogen pickup relative to generic Zircaloy-2, such that a 1 percent failure strain limit for these rods at 72 GWd/MTU is reasonable for BWRs. Additionally, due to the greatly reduced corrosion and hydrogen pickup of coated cladding, Cr-coated cladding is expected to perform better than these alloys. Likewise, FeCrAl does not exhibit hydride embrittlement and is also expected to perform better than these alloys.

However, it is likely that a greater number of fuel rods above 55 GWd/MTU will be loaded into a high burnup transport package than was assumed in the existing analysis. The NUREG/CR-6672 does not give the full details of the finite element analysis such that it could be re-performed using a different distribution of failure fractions. In lieu of this, we recommend increasing the failure fractions that were used in the existing analysis by a factor of 2.0 to account for a greater number of fuel rods with 1 percent strain capacity.

Table B-5 and Table B-6 below show the recommended failure fraction for each velocity range.

Table B-5 Recommended Failure Fraction for Each Velocity Range for Pressurized Water Reactors

Accident Velocity (mph)	Failure Fraction Original Analysis	Failure Fraction New Analysis
>90 Cases 1, 8, 9, 10, 11, 12, 13, 16, 17	1.0	1.0
60–90 Cases 5, 6, 7, 15	0.59	1.0
30–60 Cases 2, 3, 4, 14	0.25	0.5

mph = miles per hour.

Table B-6 Recommended Failure Fraction for Each Velocity Range for Boiling Water Reactors

Accident Velocity (mph)	Failure Fraction Original Analysis	Failure Fraction New Analysis
>90 Cases 1, 8, 9, 10, 11, 12, 13, 16, 17	1.0	1.0
60–90 Cases 5, 6, 7, 15	0.20	0.40
30–60 Cases 2, 3, 4, 14	0.03	0.06

mph = miles per hour.

B.6 Particulate Release

The existing analysis performs a relatively in-depth assessment to bound the particulate releases from various scenarios. This analysis derives release fractions of very small particles (<10 microns [µm]) applicable for the fire-only scenario and for the scenario with increased temperature and impact. Doped fuel typically results in larger fuel grain sizes and is not expected to negatively affect the particulate release fraction.

For high burnup fuel, the release fraction from impact only is not expected to significantly change. For example, Vlassopoulos et al. (2021-TN8679)[3] showed that following impact and bending tests, there is no more fuel release below 100 GWd/MTU than at 20 GWd/MTU, likely due to the fuel-clad bonding that occurs at higher burnup. However, it has been observed in high-temperature loss-of-coolant tests, that there is significant expulsion of material for high burnup fuel that is not observed for low burnup fuel. Because this calculation is primarily interested in small (<10 µm) particles, the maximum expected release can be bounded by assuming the entire volume of high burnup rim (10 percent of the pellet volume at 72 GWd/MTU) could break into <1 µm particles during a thermal event. Additionally, we could conservatively assume that no more than 10 percent of the fuel rods in the cask are at high burnup (>60 GWd/MTU), whereby 1 percent of the fuel could be available for release.

Using the same methodology as NUREG/CR-6672 for evaluating the burst opening and transport through a packed bed of larger particles, the release fractions in Table B-7 can be

[3] Vlassopoulos, E, Papaioannou, D, Nasyrow, R, Rondinella, V, Caruso, S, Schweitzer, E, 2021, "Experimental Study on the Mechanical stability of a 50 GWd/MTU Nuclear Fuel Rod," Proceedings of the 2021 TopFuel Meeting, Spain.

derived. The NUREG/CR-6672 methodology assumes that for the fire-only scenario the cladding rupture could be large, and that fines in up to 1 foot (ft) of the rod could escape without filtering. For the impact and temperature scenario, the cladding rupture opening is expected to be smaller and fines in up to 0.25 inches (in.) of the rod could escape without filtering.

Table B-7 Changes to Particulate Release Fractions

Accident	Release Fraction Original Analysis	Release Fraction New Analysis
Impact and temperature Cases 1–17	3.0E-5	1.5E-4
No impact, fire only Case 18	4.0E-7	9.3E-4

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.

For a no impact, fire-only scenario, assume that all the particulates in a 1 ft section will be released, and 1 percent of the remainder will be released:

$$F_{RC} = (1.0 \times 10^{-2}) \left[\frac{1}{12} + \frac{11}{12}(0.01) \right] = 9.3 \times 10^{-4}$$

F_{RC} is the fraction of the materials in a spent fuel rod that is released to the cask interior upon rod failure.

For impact and temperature, assume that all the particulates in a 0.25 in. section will be released and 1 percent of the remainder will be released. Use the 120 mph impact to bound the impact release:

$$F_{RC} = (1.0 \times 10^{-2} + 2.9 \times 10^{-3}) \left[\frac{0.25}{144} + \frac{143.75}{144}(0.01) \right] = 1.5 \times 10^{-4}$$

Table B-7 shows the changes to these release fractions. The release fractions for the fire-only scenario are greater because of the larger expected burst opening in this case and the substantially greater potential for fuel fragmentation in high burnup fuel.

B.7 Cesium and Rubidium Release

The existing analysis calculates upper bound release fractions for both Cs and Ru for the fire-only scenario (case 18), the impact that results in a long engulfing fire (cases 4, 7, 10, 13), and the events that result in fuel oxidation (cases 14, 15, 16, 17). For all other cases, it was determined that the temperature is not great enough to result in additional Cs or Ru releases.

For spent fuel rods at 72 GWd/MTU, the quantities of Cs and Ru will be greater, but there is no credible mechanism that manifests between 60 and 72 GWd/MTU that would challenge the conservative approach used in NUREG/CR-6672 to determine release fractions. Likewise, the use of doped fuel is not expected to affect these release fractions and is sufficiently covered by the conservatisms applied in this analysis. NUREG/CR-6672 conservatively assumes that all of the Cs is released to the pellet surfaces and then calculates release fractions based on the vapor pressures of likely Cs chemical species using the VICTORIA code. The impact of burnup on these likely chemical species is small relative to the assumption that all the Cs is released to the pellet surface. Therefore, for this analysis, we retain the previous release fractions of Cs and Ru, as shown in Table B-8.

Table B-8 Cesium and Rubidium Release Fractions for Both Analyses

Category	Case Number	Cesium Release Fraction	Rubidium Release Fraction
Impact events that initiate hot, engulfing, optically dense, long-duration fires	4, 7, 10, 13	5.0E-5	3.0E-5
Fire only	18	2.0E-5	1.3E-4
Events that result in fuel oxidation	14, 15, 16, 17	1.5E-4	4.0E-7
All other events	1, 2, 3, 4, 5, 8, 9, 11, 12	0.0	0.0

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
The Case Numbers mentioned in this table have been taken from NUREG/CR-6672, § 7.2.6 (Sprung et al. 2000-TN222), Accident Cases and the number may vary between the two analyses.

B.8 Xenon and Krypton Release

The existing analysis assumes 100 percent release of noble gas for all cases (both fire and impact) and therefore does not take credit for any pellet retention of Xe or Kr, which is typically around 95 percent for fuel rods below 60 GWd/MTU and could be reduced to 80 percent for higher burnup rods. This assessment is bounding for all fuel types, including doped fuel. For this calculation, the existing Kr parameters are retained.

B.9 Crud Release

Increased burnup is not expected to lead to the formation of any additional crud[4] or make the crud more susceptible to being released from the fuel-cladding. In fact, modern LWRs operate with improved coolant chemistry controls that result in lower crud formation than was observed 20 years ago. For this calculation, the existing crud parameters are retained. The mechanisms behind crud formation are not well known and it is possible that the introduction of ATF cladding may change the rate of crud formation either due to a difference in surface roughness or surface chemistry. If these issues come up, industry may alter manufacturing parameters or coolant chemistry to mitigate them. In the long-term increased crud is not expected, but there may be some transition batches with higher crud thicknesses.

B.10 Items Changed

The following items would be changed in the RADTRAN input for PWR fuel at 72 GWd/MTU relative to the existing analysis:

- radionuclides in assembly
- expansion factors
- rod failure fractions
- particulate release

B.11 New Values

Table B-9 and Table B-10 provide the updated release fractions from the RADTRAN analyses and assessments described above. The cases refer to the 18 events (cases) for PWR and BWR fuel rods analyzed in NUREG/CR-6672.

[4] A colloquial term for corrosion and wear products (rust particles, etc.) that become radioactive (i.e., activated) when exposed to radiation.

Table B-9 New Release Fractions for 72 GWd/MTU for Pressurized Water Reactors

Case	Krypton	Cesium	Rubidium	Particulates	Crud
1	8.16E-01	2.45E-08	6.12E-07	3.06E-06	2.04E-03
2	3.47E-01	1.04E-08	2.60E-07	1.30E-06	1.73E-03
3	4.07E-01	1.22E-08	3.05E-07	1.52E-06	2.03E-03
4	8.41E-01	2.93E-05	2.55E-06	1.28E-05	3.11E-03
5	8.16E-01	2.45E-08	6.12E-07	3.06E-06	2.04E-03
6	8.88E-01	2.66E-08	6.66E-07	3.33E-06	2.22E-03
7	9.10E-01	5.90E-06	6.82E-07	3.41E-06	2.47E-03
8	9.43E-01	2.83E-08	7.07E-07	3.54E-06	2.36E-03
9	9.65E-01	2.90E-08	7.24E-07	3.62E-06	2.41E-03
10	9.72E-01	5.90E-06	7.29E-07	3.65E-06	2.63E-03
11	9.43E-01	2.83E-08	7.07E-07	3.54E-06	2.36E-03
12	9.65E-01	2.90E-08	7.24E-07	3.62E-06	2.41E-03
13	9.72E-01	5.90E-06	7.29E-07	3.65E-06	2.63E-03
14	9.49E-01	8.15E-05	7.24E-05	6.97E-05	7.01E-03
15	9.72E-01	5.90E-06	6.46E-06	3.65E-06	3.41E-03
16	9.72E-01	5.90E-06	6.46E-06	3.65E-06	3.41E-03
17	9.72E-01	5.90E-06	6.46E-06	3.65E-06	3.41E-03
18	8.39E-01	1.68E-05	6.71E-08	1.56E-04	2.52E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.

Table B-10 New Release Fractions for 72 GWd/MTU for Boiling Water Reactors

Case	Krypton	Cesium	Rubidium	Particulates	Crud
1	8.16E-01	2.45E-08	6.12E-07	3.06E-06	2.04E-03
2	1.55E-02	3.22E-10	8.06E-09	4.03E-08	4.48E-04
3	3.00E-02	9.00E-10	2.25E-08	1.13E-07	1.25E-03
4	8.36E-01	4.06E-05	4.73E-06	2.36E-05	3.12E-03
5	2.58E-01	7.75E-09	1.94E-07	9.69E-07	1.62E-03
6	3.14E-01	9.41E-09	2.35E-07	1.18E-06	1.96E-03
7	8.38E-01	3.21E-05	3.04E-06	1.52E-05	3.14E-03
8	8.16E-01	2.45E-08	6.12E-07	3.06E-06	2.04E-03
9	8.88E-01	2.66E-08	6.66E-07	3.33E-06	2.22E-03
10	9.10E-01	5.90E-06	6.82E-07	3.41E-06	2.47E-03
11	8.16E-01	2.45E-08	6.12E-07	3.06E-06	2.04E-03
12	8.88E-01	2.66E-08	6.66E-07	3.33E-06	2.22E-03
13	9.10E-01	5.90E-06	6.82E-07	3.41E-06	2.47E-03
14	8.37E-01	1.19E-04	1.03E-04	1.17E-04	6.46E-03
15	8.38E-01	7.79E-05	6.88E-05	7.02E-05	6.19E-03
16	9.10E-01	5.90E-06	6.42E-06	3.41E-06	3.25E-03
17	9.10E-01	5.90E-06	6.42E-06	3.41E-06	3.25E-03
18	8.39E-01	1.68E-05	6.71E-08	1.56E-04	2.52E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.

B.12 85 GWd/MTU

The biggest change due to increasing from 72 GWd/MTU to 85 GWd/MTU beyond the radionuclide production rates would be the particulate release fraction. Using the same methodology as previously described where the volume of the pellet that has transformed to the high burnup rim structure is available as particulate release, in going from 72 to 85 GWd/MTU, the available volume increases from 10 percent to 20 percent and the release fractions go up, as seen in Table B-11.

Table B-11 Changes to Particulate Release Fractions

Accident	Rod to Cask Release Fraction Original Analysis	Rod to Cask Release Fraction 72 GWd/MTU	Rod to Cask Release Fraction 85 GWd/MTU
Impact and temperature Cases 1–17	3.0E-5	1.5E-4	2.7E-4
No impact, fire only Case 18	4.0E-7	9.3E-4	1.8E-3

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
GWd/MTU = gigawatt days (units of energy) per metric ton uranium.

For this case, the new values for 85 GWd/MTU are those listed in Table B-12 and Table B-13 for PWRs and BWRs, respectively.

Table B-12 New Release Fractions for 85 GWd/MTU for Pressurized Water Reactors

Case	Krypton	Cesium	Rubidium	Particulates	Crud
1	8.16E-01	2.45E-08	6.12E-07	5.51E-06	2.04E-03
2	3.47E-01	1.04E-08	2.60E-07	2.34E-06	1.73E-03
3	4.07E-01	1.22E-08	3.05E-07	2.74E-06	2.03E-03
4	8.41E-01	2.93E-05	2.55E-06	2.30E-05	3.11E-03
5	8.16E-01	2.45E-08	6.12E-07	5.51E-06	2.04E-03
6	8.88E-01	2.66E-08	6.66E-07	5.99E-06	2.22E-03
7	9.10E-01	5.90E-06	6.82E-07	6.14E-06	2.47E-03
8	9.43E-01	2.83E-08	7.07E-07	6.37E-06	2.36E-03
9	9.65E-01	2.90E-08	7.24E-07	6.52E-06	2.41E-03
10	9.72E-01	5.90E-06	7.29E-07	6.56E-06	2.63E-03
11	9.43E-01	2.83E-08	7.07E-07	6.37E-06	2.36E-03
12	9.65E-01	2.90E-08	7.24E-07	6.52E-06	2.41E-03
13	9.72E-01	5.90E-06	7.29E-07	6.56E-06	2.63E-03
14	9.49E-01	8.15E-05	7.24E-05	1.26E-04	7.01E-03
15	9.72E-01	5.90E-06	6.46E-06	6.56E-06	3.41E-03
16	9.72E-01	5.90E-06	6.46E-06	6.56E-06	3.41E-03
17	9.72E-01	5.90E-06	6.46E-06	6.56E-06	3.41E-03
18	8.39E-01	1.68E-05	6.71E-08	3.02E-04	2.52E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.

Table B-13 New Release Fractions for 85 GWd/MTU for Boiling Water Reactors

Case	Krypton	Cesium	Rubidium	Particulates	Crud
1	8.16E-01	2.45E-08	6.12E-07	5.51E-06	2.04E-03
2	1.55E-02	3.22E-10	8.06E-09	7.25E-08	4.48E-04
3	3.00E-02	9.00E-10	2.25E-08	2.03E-07	1.25E-03
4	8.36E-01	4.06E-05	4.73E-06	4.26E-05	3.12E-03
5	2.58E-01	7.75E-09	1.94E-07	1.74E-06	1.62E-03
6	3.14E-01	9.41E-09	2.35E-07	2.12E-06	1.96E-03
7	8.38E-01	3.21E-05	3.04E-06	2.73E-05	3.14E-03
8	8.16E-01	2.45E-08	6.12E-07	5.51E-06	2.04E-03
9	8.88E-01	2.66E-08	6.66E-07	5.99E-06	2.22E-03
10	9.10E-01	5.90E-06	6.82E-07	6.14E-06	2.47E-03
11	8.16E-01	2.45E-08	6.12E-07	5.51E-06	2.04E-03
12	8.88E-01	2.66E-08	6.66E-07	5.99E-06	2.22E-03
13	9.10E-01	5.90E-06	6.82E-07	6.14E-06	2.47E-03
14	8.37E-01	1.19E-04	1.03E-04	2.11E-04	6.46E-03
15	8.38E-01	7.79E-05	6.88E-05	1.26E-04	6.19E-03
16	9.10E-01	5.90E-06	6.42E-06	6.14E-06	3.25E-03
17	9.10E-01	5.90E-06	6.42E-06	6.14E-06	3.25E-03
18	8.39E-01	1.68E-05	6.71E-08	3.02E-04	2.52E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.

B.13 References

Hall, R., R. Sweet, R. Belles, and W.A. Wieselquist. 2021. *Extended-Enrichment Accident-Tolerant LWR Fuel Isotopic and Lattice Parameter Trends*. ORNL/TM-2021/1961, Oak Ridge, Tennessee. March. ADAMS Accession No. ML21088A254. TN8286.

Kucinski, N., P. Stefanovic, J. Clarity, and W. Wieselquist. 2022. Impacts of LEU+ and HBU Fuel on Decay Heat and Radiation Source Term. ORNL/TM-2022/1841, Oak Ridge, Tennessee. ADAMS Accession No. ML22159A191. TN8091.

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Vlassopoulos, E., D. Papaioannou, R. Nasyrow, V. Rondinella, S. Caruso, and E.W. Schweitzer. 2021. *Experimental Study on the Mechanical Stability of a 59 GWd/tHM Nuclear Fuel Rod*. Proceedings of the 2021 TopFuel Meeting, Santander, Spain. TN8679.

APPENDIX C

SITE AND ROUTE SELECTION

Spent nuclear fuel (SNF) and high-level radioactive waste are currently stored at 77 locations in the United States (67 nuclear power plant [NPPs], five storage facilities at sites of decommissioned NPPs, and five U.S Department of Energy [DOE] defense facilities). The U.S. Nuclear Regulatory Commission (NRC) selected six NPP sites—at least one for each region of the United States (see Table C-1)— upon which to base the performance of a generic (e.g., bounding) analysis of the environmental effects of the transportation of accident tolerant fuel (ATF). The sites were chosen based on their inclusion in Ramsdell et al. (2001-TN4545) as example sites for transportation analysis. Dresden Generating Station (Dresden) NPP was selected over the previous Zion site because the Zion NPP was decommissioned. Spent fuel transportation routes were selected based on each NPP site shipping to a surrogate geologic repository. The proposed Yucca Mountain geologic repository site was used in this study as the surrogate geologic repository based on the Nuclear Waste Policy Act and past DOE and NRC transportation studies. A surrogate destination is used for this analysis to bound the transportation impacts of SNF because no active geologic repository site is currently available. Three nuclear fuel fabrication facilities provide unirradiated light-water reactor (LWR) fuel assemblies, and each of them is expected to manufacture ATF. Since two nuclear fuel fabrication facilities are in the eastern half of the United States along with most of the selected NPPs, one unirradiated fuel shipment route was selected as a representative, or bounding, route based on the longest route from a nuclear fuel fabrication facility to one of the six NPP sites. This would be a route from the Framatome, Inc. Fuel Fabrication Facility (Framatome FFF) in Richland, Washington, to the Turkey Point Nuclear Generating Station (Turkey Point) NPP located near Homestead, Florida. Table C-2 lists the routes modeled.

The routing code Web-Based Transportation Routing Analysis Geographic Information System (WebTRAGIS) software (Peterson 2018-TN5839) provides the necessary routing information that can be directly imported into NRC-Radioactive Material Transport (RADTRAN), such as the one-way distance and the populations within 800 meters (m; ½ mi) of a selected route. Both truck and rail route information can be provided by WebTRAGIS and are used in this study for illustrative purposes. No actual spent fuel shipments on these routes are occurring or planned. WebTRAGIS determines routes from specified starting and ending points for highway, rail, or waterway transportation within the continental United States and provides the necessary information for each State traversed by a particular route. Routes are broken into “links,” or smaller segments of highway, railway, or waterway. WebTRAGIS derives route information around each network link along the transportation route, where link population densities and route distances are reported by rural, suburban, and urban categories. Various criteria for the route(s) to be determined may be specified, such as Highway Route Controlled Quantity criteria, which are used for the SNF truck routes presented within this document. WebTRAGIS also has a setting for hazardous material (HAZMAT) transportation because certain routes are unavailable to vehicles carrying HAZMAT. Nuclear fuel, regardless of whether it has been irradiated, is considered HAZMAT and therefore HAZMAT transportation settings were enabled.

As was performed in NUREG/CR-6703 (Ramsdell et al. 2001-TN4545), incident-free legal weight truck transportation of spent ATF is evaluated by considering shipments from six representative reactor sites to the surrogate Yucca Mountain, Nevada, geologic repository for disposal. This assumption is conservative because it tends to maximize the shipping distance from the East Coast and the Midwest where most of the NPPs are located. A rail shipment of spent ATF was evaluated as a sensitivity case for a single reactor site in the Northeast. Representative reactor sites in each NRC region were selected to illustrate the impacts of transporting spent ATF from a variety of possible locations. These regions and the representative NPPs are listed in Table C-1.

Table C-1 Sites Used for Transportation Evaluation

Nuclear Power Plant (NPP)	Represented Region
Turkey Point NPP	Region II
Brunswick NPP	Region II
Millstone NPP	Region I
Fermi NPP	Region III
Dresden NPP	Region III
Columbia NPP	Region IV

Turkey Point = Turkey Point Nuclear Generating Station, Brunswick = Brunswick Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, Fermi = Enrico Fermi Nuclear Generating Station, Dresden = Dresden Generating Station, Columbia = Columbia Generating Station.

Route distance information for the transportation of irradiated ATF (i.e., spent ATF) from each reactor site to the surrogate high-level waste repository at Yucca Mountain is listed in Table C-2. Of these transportation routes, the longest one-way distance from a reactor site to Yucca Mountain is the route from Millstone, Connecticut. The routes with the longest distances through urban areas are the routes from Millstone and from Dresden, Illinois. The routes with the largest amount of transit through suburban areas are from Millstone and from Turkey Point, Florida.

Table C-2 Shipping Distances

Origin Site	Mode	One-Way Shipping Distance (km)(a)(b)	Rural Distance (km)(a)	Suburban Distance (km)(a)	Urban Distance (km)(a)
Framatome FFF	Truck	5,129	3,786	1,184	160
Brunswick	Truck	3,982	2,984	904	94
Columbia	Truck	1,461	1,387	73	0.3
Dresden	Truck	2,965	2,542	375	48
Fermi	Truck	3,428	2,786	578	65
Millstone	Truck	4,457	3,387	935	134
Turkey Point	Truck	4,251	3,151	915	185
Brunswick	Rail	4,843	3,491	1,187	165
Columbia	Rail	1,960	1,659	253	48
Dresden	Rail	3,111	2,507	535	69
Fermi	Rail	3,756	2,794	785	177
Millstone	Rail	4,787	3,312	1,248	227
Turkey Point	Rail	5,328	3,813	1,276	239

Framatome FFF = Framatome, Inc. Fuel Fabrication Facility, Brunswick = Brunswick Nuclear Generating Station, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station Millstone = Millstone Nuclear Power Plant, Turkey Point = Turkey Point Nuclear Generating Station.

(a) To convert kilometer (km) to mile (mi), multiply by 0.621371.

(b) One-way shipping distances for the listed nuclear power plants (NPPs) is to the surrogate geologic repository Yucca Mountain site and the one-way shipping distance from the Framatome FFF is to the Turkey Point NPP site.

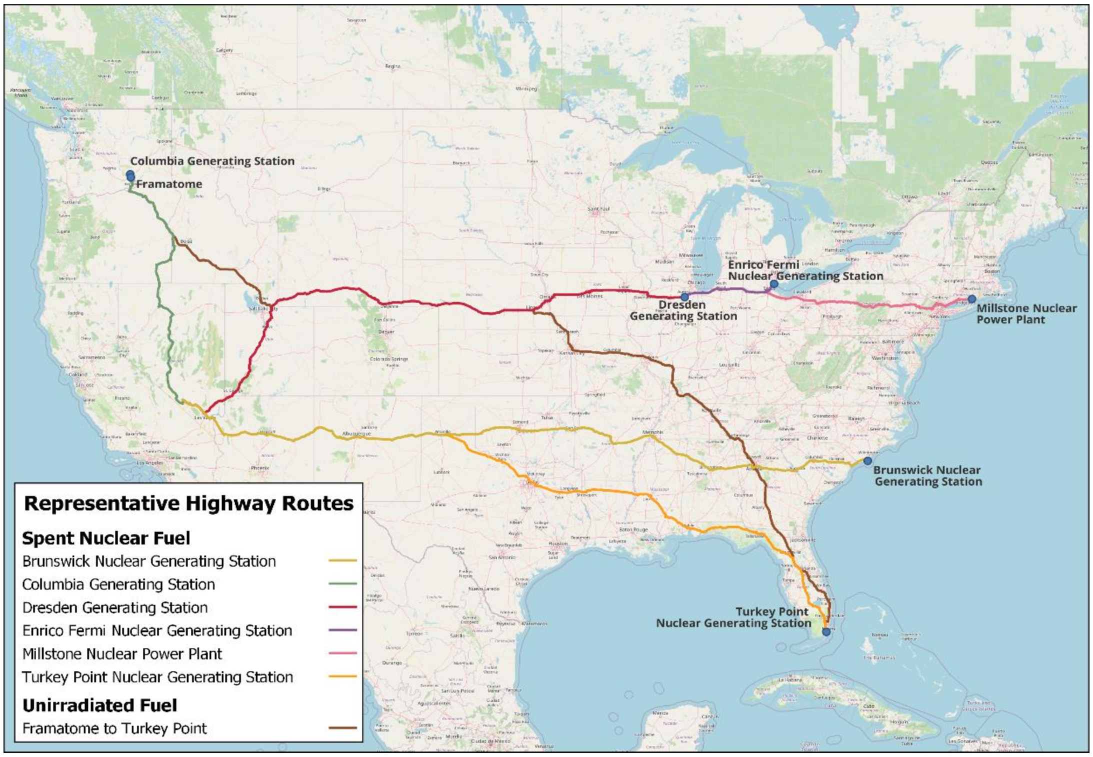

Figure C-1 Highway Routes Across the United States

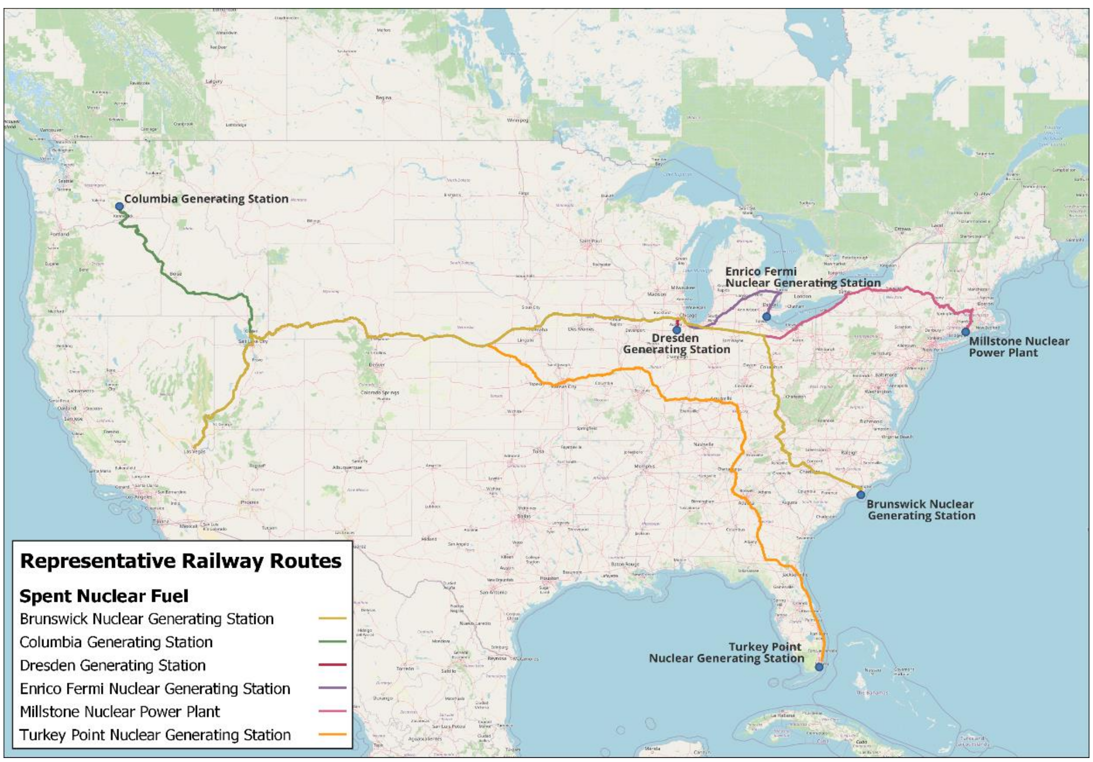

Figure C-2 Rail Routes Across the United States

C.1 References

Peterson, S. 2018. *WebTRAGIS: Transportation Routing Analysis Geographic System User's Manual*. ORNL/TM-2018/856, Oak Ridge National Laboratory, Oak Ridge, Tennessee. ADAMS Accession No. ML18324A611. TN5839.

Ramsdell, J.V. Jr., C.E. Beyer, D.D. Lanning, U.P. Jenquin, R.A. Schwarz, D.L. Strenge, P.M. Daling, and R.T. Dahowski. 2001. *Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU*. NUREG/CR-6703, Pacific Northwest National Laboratory, Richland, Washington. ADAMS Accession No. ML010310298. TN4545.

APPENDIX D

DATA AND PARAMETER VALUES FOR TRANSPORTATION EVALUATION

This appendix provides the input parameter values, reference sources, and additional information concerning the inputs for the radiological impact calculations using the U.S. Nuclear Regulatory Commission-Radioactive Material Transport (NRC-RADTRAN) code and data applied for the nonradiological accident impacts (Statement of Web-Based Transportation Routing Analysis Geographic Information System [WebTRAGIS], RADTRAN manuals, and publicly available databases as source of information). For example, some vehicle input parameter values were obtained from the Federal Motor Carrier Safety Administration (FMCSA) data sources.

D.1 NRC-RADTRAN Transportation Input Parameter Values

The information in Table D-1 through Table D-7 is listed by the input tabs in the NRC-RADTRAN graphical user interface (GUI) (Ball and Zavisca 2020-TN8073). The Loss of Shielding Tab, Economic Model Tab, and Default Parameters Tab are not applied, so they are not reflected in the following series of tables.

Table D-1 NRC-RADTRAN Transportation Input Parameter Values for the Vehicles Tab

Input Parameter	Value with Units	Reference Source	Comments
Name	VEHICLE_1	--	User specified
Transport Mode	Highway (Rail)	--	Truck transport with rail transport as a sensitivity calculation
Exclusive Use	Yes	Radioactive Material Transport (RADTRAN) 6 Technical Manual (Weiner et al. 2014-TN3389)	Maximizes external dose rate to the regulatory limits of 10 CFR 71.47 (TN301)
Size (CD)	5.08 m	NAC International-Legal Weight Truck (NAC-LWT) (Docket No. 71-9225)	Steel-Lead-Steel package used in NUREG/CR-6672
Dose Rate at 1 m	14	N/A	Exclusive use will set the external dose rate
Gamma Fraction	1.0	Maheras et al. 2023-TN8104	--
Neutron Fraction	0.0	Maheras et al. 2023-TN8104	--
Crew Size	2	AEC 1972 (TN22); NRC 1977 (TN417); DOE 2002 (TN418)	Crew size for truck transportation
Crew Distance	3.5 m	NUREG-2125, Table B-1 (NRC 2014-TN3231)	While for a different package, location on trailer similar to that expected for NAC-LWT package
Width Facing Crew	1.12 m	NAC-LWT CoC (Docket No. 71-9225)	Steel-Lead-Steel package used in NUREG/CR-6672 (Sprung et al. 2000-TN222)
Crew Shielding	1	NUREG/CR-6672 (Sprung et al. 2000-TN222)	No shielding to maximize crew dose
Number of Shipments	1	--	Unit shipment assessment

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All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073).

Table D-2 NRC-RADTRAN Transportation Input Parameter Values for the Links Tab

Input Parameter	Value with Units	Reference Source	Comments
Name	[State]_[Population Type]_[Mode]	Web-Based Transportation Routing Analysis Geographic Information System (WebTRAGIS)	A WebTRAGIS provided value
Vehicle	VEHICLE_1	--	See VEHICLE inputs
Mode	Primary Highway (Nonroad)	Online WebTRAGIS	A WebTRAGIS provided value. Nonroad for rail transport sensitivity
Length	[by each link]	Online WebTRAGIS	A WebTRAGIS provided value
Speed	[by each link]	Online WebTRAGIS	See Table D-1
Adjacent Vehicle Occupants	2	DOE 2002 (TN418)	Rounded up from 1.5
Population Density	[by each link]	Online WebTRAGIS	A WebTRAGIS provided value with adjustments to current Census data. See Table D-8 through Table D-14
Traffic	[by each State]	RADTRAN 6/RADCAT 6.0 User Guide (Weiner et al. 2013-TN3390)	See Table D-15
Accidents per km	[by each State]	Federal Motor Carrier Safety Administration (FMCSA) website	FMCSA 2022-TN8075 for large trucks for the year 2021. See Table D-16
Deaths per Accident	[by each State]	FMCSA website	FMCSA 2022-TN8075 for large trucks for the year 2021. See Table D-16
Population Type	[Rural, Suburban, or Urban]	Online WebTRAGIS	A WebTRAGIS provided value
Farm Fraction if Rural	0.5 if Rural, 0 else	Online WebTRAGIS	A WebTRAGIS provided value

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All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User’s Guide (Ball and Zavisca 2020-TN8073).

Table D-3 NRC-RADTRAN Transportation Input Parameter Values for the Stops Tab

Input Parameter	Value with Units	Reference Source	Comments
Name	STOP_1/STOP_2	--	--
Vehicle	VEHICLE_1	--	--
Population Density	30,000/340 people/km^2	NUREG/CR-6672 (Sprung et al. 2000-TN222)	30,000 people/km^2 based on nine persons within 10 m of vehicle.
Inner Radius	1/10 m	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Min/max radii of annular area around vehicle at stops
Outer Radius	10/800 m	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Min/max radius of annular area surrounding truck stop
Shielding Factor	1/0.2	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Inner/Outer radius shielding factor applied to annular area surrounding vehicle at stops
Duration	0.3/0.3 h	Griego et al. (Griego et al. 1996-TN69)	Based on one 18-minute stop per 4-hour driving time from the.

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All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073).

Table D-4 NRC-RADTRAN Transportation Input Parameter Values for the Handling Tab

Input Parameter	Value with Units	Reference Source	Comments
Name	HANDLE_1	--	--
Vehicle	VEHICLE_1	--	--
Persons	5	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Table 3.3 states number of handlers has been updated based on recent empirical data.
Distance	1 m	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Table 3.3 states that value is based on empirical data that confirm original NUREG-0170 value.
Duration	0.5 h	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Table 3.3 states that value is based on empirical data that confirm original NUREG-0170 value.

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All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073).

Table D-5   NRC-RADTRAN Transportation Input Parameter Values for the Packages Tab

Input Parameter	Value with Units	Reference Source	Comments
Name	Package_1	--	--
Largest (critical) Dimension	From Vehicle_1 input	--	--
Dose Rate at 1 m from surface	From Vehicle_1 input	--	--
Gamma Fraction	From Vehicle_1 input	--	--
Neutron Fraction	From Vehicle_1 input	--	--

-- = no content in the table cell.
All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073).

Table D-6   NRC-RADTRAN Transportation Input Parameter Values for the Accidents Tab

Input Parameter	Value with Units	Reference Source	Comments
Severity Probabilities by mode and population group	Various	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Table 7.31 for truck packages
Release Fractions by Release Groups	Various	NUREG/CR-6672 (Sprung et al. 2000-TN222)	Table 7.31 for pressurized water reactor (PWR) and boiling water reactor (BWR) Steel-Lead-Steel packages. Sensitivity calculations based on information in Appendix B
Weather	National Average	RADTRAN 6 Technical Manual (Weiner et al. 2014-TN3389) and NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073)	"National Average" value requires no other inputs
Isopleths (Dispersion Areas)	Select "From Links table"	RADTRAN 6 Technical Manual (Weiner et al. 2014-TN3389) and NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073)	Normally, all isopleths use the same population density (taken from the Link where the accident occurs).

All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073).

Table D-7 NRC-RADTRAN Transportation Input Parameter Values for the Radionuclides Tab

Input Parameter	Value with Units	Reference Source	Comments
Package Name	Package_1	--	--
Isotope	Based on A2 values with Kr gas and Crud (Co-60)	Oak Ridge National Laboratory (ORNL) Phase I (Hall et al. 2021-TN8084 and Cumberland et al. 2021-TN8085) and Phase 2 (Kucinski et al. 2022-TN8091) reports	See Appendix A
Release Group	Particulate, Cu, Ru, Gas, or Crud	--	See Appendix A
Inventory	Various	ORNL Phase I (Hall et al. 2021-TN8084 and Cumberland et al. 2021-TN8085) and Phase 2 (Kucinski et al. 2022-TN8091) reports	See Appendix A

-- = no content in the table cell.
All values are kept as provided in the NRC-RADTRAN installation package and discussed in the NRC-RADTRAN 1.0 Quick Start User's Guide (Ball and Zavisca 2020-TN8073).

D.2 Truck and Rail Accident Rates

FMCSA publishes information through the Motor Carrier Management Information System. The summary of statistics for large trucks pertaining to the number of truck crashes, number fatal crashes, and number injury crashes due to trucks travel by State for calendar year 2021 were obtained from FMCSA's Analysis and Information Online database website at https://ai.fmcsa.dot.gov/CrashStatistics/rptSummary.aspx. Using these data along with associated total miles traveled by trucks in a State, the truck accident rate for each State is determined and used in the RADTRAN analysis for the segment of the route falling in the respective State from origin to destination.

The rail accident rate is determined based on the paper by Development of Rail Accident Rates for Spent Nuclear Fuel Rail Shipments-17088 (Abkowitz and Bickford 2017-TN8101) using the equation:

Rail Accident Rate (per mile) = train-mile accident rate per mile + [(car-mile accident rate per mile) × (number of cars in train)]

D.3 Annual Number of Accident Tolerant Fuel Shipments

Unirradiated accident tolerant fuel (ATF):

For one-third-core reloads:

- Pressurized water reactor (PWR) (WASH-1238): The reference LWR is approximately 1,100 MWe gross PWR with 60 fuel assemblies per core reload. There can be 10 PWR unirradiated fuel assemblies per shipment in 10 Traveller packages (see Figure 3-7). With

2-year refueling frequencies, this means there are approximately 3 PWR unirradiated ATF shipments per year (60/2 = 30 assemblies per year; 30/10 assemblies per shipment = 3 shipments per year).

- Boiling water reactor (BWR) (Constellation-TN8102): An approximately 1,100 MWe gross BWR-6 plant (similar to the reference plant in WASH-1238) has 208 fuel assemblies per core reload. Based on the weight limit for a freight truck, there could be up to 28 BWR unirradiated fuel assemblies per shipment in 14 packages where there are 2 BWR assemblies in a RAJ-II package (Figure 3-8). Thus, with 2-year refueling frequencies, this means there are approximately 4 BWR unirradiated fuel shipments per year (208/2 = 104 assemblies per year; 104/28 assemblies per shipment = 3.71 shipments per year rounded up to 4 shipments per year).

Spent ATF:

- The shipment numbers of spent ATF assemblies would be the number transferred from the reactor core that coincides with the number of unirradiated ATF assemblies needed to support the core reloads mentioned above for unirradiated ATF shipments.
- PWR: Based on the analysis in WASH-1238, one spent ATF assembly per package and one package per shipment. Thus, 60 shipments over 2 years means 60 spent ATF assemblies per reload/2 years between reloads/1 spent ATF PWR assembly per package equals 30 PWR spent ATF shipments per year.
- BWR: Two spent ATF assemblies per package, one package per shipment. Thus, 208 spent ATF assemblies per reload/2 years between reloads/2 spent ATF BWR assemblies per package equals approximately 52 spent ATF BWR shipments per year.

For half-core reloads:

- Pressurized water reactor (PWR) (WASH-1238): Scaling for half-core reloads, the reference LWR is approximately 1,100 MWe gross PWR and would have 90 fuel assemblies per half-core reload. There can be 10 PWR unirradiated fuel assemblies per shipment in 10 Traveller packages (see Figure 3-7). With 2-year refueling frequencies, this means there are approximately 5 PWR unirradiated ATF shipments per year (90/2 = 45 assemblies per year; 45/10 assemblies per shipment = 4.5 shipments per year rounded up to 5 shipments per year).
- Boiling water reactor (BWR) (Constellation-TN8102): Likewise, scaling for half-core reloads, an approximately 1,100 MWe gross BWR-6 plant (similar to the reference plant in WASH-1238) would have 312 fuel assemblies per core reload. Based on the weight limit for a freight truck, there could be up to 28 BWR unirradiated fuel assemblies per shipment in 14 packages where there are 2 BWR assemblies in a RAJ-II package (Figure 3-8). Thus, with 2-year refueling frequencies, this means there are approximately 6 BWR unirradiated fuel shipments per year (312/2 = 156 assemblies per year; 156/28 assemblies per shipment = 5.57 shipments per year rounded up to 6 shipments per year).

Spent ATF:

- The shipment numbers of spent ATF assemblies would be the number transferred from the reactor core that coincides with the number of unirradiated ATF assemblies needed to support the half-core reloads mentioned above for unirradiated ATF shipments.
- PWR: Based on the analysis in WASH-1238, scaled up to half-core reloads, one spent ATF assembly per package and one package per shipment. Thus, 90 shipments over 2 years

means 90 spent ATF assemblies per reload/2 years between reloads/1 spent ATF PWR assembly per package equals 45 PWR spent ATF shipments per year.

- BWR: Similarly, two spent ATF assemblies per package, one package per shipment. Thus, 312 spent ATF assemblies per reload/2 years between reloads/2 spent ATF BWR assemblies per package equals approximately 78 spent ATF BWR shipments per year.

D.4 Population Density Adjustments

The population datasets used by WebTRAGIS were developed from a combination of data sources, including 2010 U.S. Census Bureau block group population data, American Community Survey intercensal data, U.S. Census Bureau's Topologically Integrated Geographic Encoding and Referencing (TIGER) system road data, slope from the National Imagery and Mapping Agency's (NIMA's) Digital Terrain Elevation Data, and land cover from the United States Geological Survey National Land Cover Database (Peterson 2018-TN5839). The year of the population density data as provided in the WebTRAGIS output file RouteDensityByState.csv is stated as 2012. To account for the changes in population density since 2012 based on the 2020 U.S. Census data for a current year, this appendix provides the population density adjustments to the time of the NRC-RADTRAN calculations, namely for the year of 2022. First, a state population density correction factor for the year 2022 is determined based on a state's average population density for the 2010 and 2020 U.S. Census and the land area as shown in Table D-8. Then for each route, a state's population density correction factor is applied to the rural, suburban, and urban population densities along that route. This results in the corrected route population densities for each truck route shown in Table D-8 through Table D-14. Please note WebTragis provides population densities in persons per mile squared, but for use in NRC-RADTRAN the data units are converted to persons per kilometer squared.

Table D-8 Compilation of 2010 and 2020 U.S. Census Data by State to Determine Annual Average Growth Rate for the Period

State	2010 Census Data	2020 Census Data	Area (km^2)[(a)]	Average Density 2010 (per km^2)[(a)]	Average Density 2020 (per km^2)[(a)]	Percent Change in Density per Year	Change in Density for 2022 10-year Change
Alabama	4,779,736	4,893,000	135,760	35	36	0.2857	1.029
Arizona	6,392,017	7,174,000	295,000	21	24	1.4286	1.143
Arkansas	2,915,918	3,012,000	137,754	21	21	0.0000	1.000
California	37,253,956	39,538,223	403,294	92	98	0.6522	1.065
Colorado	5,029,191	5,773,714	268,317	18	21	1.6667	1.167
Connecticut	3,574,017	3,571,000	13,023	274	274	0.0000	1.000
Delaware	897,934	989,948	5,044	178	196	1.0112	1.101
Florida	18,801,310	21,220,000	170,310	110	124	1.2727	1.127
Georgia	9,687,653	10,520,000	153,909	62	68	0.9677	1.097
Idaho	1,600,000	1,754,000	216,443	7	8	1.4286	1.143
Illinois	12,830,632	12,720,000	150,010	85	84	-0.1176	0.988
Indiana	6,483,802	6,697,000	94,320	68	71	0.4412	1.044
Iowa	3,046,355	3,150,000	145,752	20	21	0.5000	1.050
Kansas	2,853,118	2,937,880	211,663	13	13	0.0000	1.000
Kentucky	4,339,367	4,505,836	102,239	42	44	0.4762	1.048
Louisiana	4,533,372	4,665,000	135,382	33	34	0.3030	1.030
Massachusetts	6,547,629	7,029,917	201,996	32	34	0.6250	1.063
Michigan	9,883,640	9,974,000	250,000	39	39	0.0000	1.000
Mississippi	2,967,297	2,982,000	123,514	24	24	0.0000	1.000
Missouri	5,988,927	6,154,913	177,976	33	34	0.3030	1.030
Nebraska	1,826,342	1,924,000	200,000	9	9	0.0000	1.000
Nevada	2,700,551	3,030,000	286,382	9	10	1.1111	1.111
New Jersey	8,791,894	8,885,000	22,610	388	392	0.1031	1.010
New Mexico	2,059,179	2,097,000	314,900	6	6	0.0000	1.000
New York	19,387,102	19,570,000	141,300	137	138	0.0730	1.007

Table D-8 Compilation of 2010 and 2020 U.S. Census Data by State to Determine Annual Average Growth Rate for the Period (Continued)

State	2010 Census Data	2020 Census Data	Area (km^2)[(a)]	Average Density 2010 (per km^2)[(a)]	Average Density 2020 (per km^2)[(a)]	Percent Change in Density per Year	Change in Density for 2022 10-year Change
North Carolina	9,535,483	10,390,000	139,390	68	74	0.8824	1.088
Ohio	11,536,504	11,680,000	116,096	99	100	0.1010	1.010
Oklahoma	3,751,351	3,949,000	181,040	20	21	0.5000	1.050
Oregon	3,831,074	4,176,000	254,810	15	16	0.6667	1.067
Pennsylvania	12,702,379	12,790,000	119,283	106	107	0.0943	1.009
South Carolina	4,625,364	5,118,425	82,932	55	61	1.0909	1.109
Tennessee	6,346,105	6,772,000	109,247	58	61	0.5172	1.052
Texas	25,145,561	29,145,505	695,662	36	41	1.3889	1.139
Utah	2,763,855	3,151,000	219,890	12	14	1.6667	1.167
Virginia	8,001,024	8,631,393	102,215	78	84	0.7692	1.077
Washington	6,724,540	7,512,000	184,830	36	40	1.1111	1.111
Wyoming	563,626	581,348	253,340	2	2	0.0000	1.000

(a) To convert km^2 to mi^2, multiply by 0.386102. Population density is reported in WebTRAGIS in mi^2; this table provides the conversion to km^2.

Table D-9 Brunswick Steam Electric Plant Truck Route Population Density

State	Rural Density/mi^2	Suburban Density/mi^2	Urban Density/mi^2	Population Correction Factor	Corrected Rural Population Density/km^2[(a)]	Corrected Suburban Population Density/km^2[(a)]	Corrected Urban Population Density/km^2[(a)]
Alabama	46.7	1,289	5,962.9	1.029	18.6	512.1	2,369.0
Arkansas	43.5	925.1	3,924.9	1.000	16.8	357.2	1,515.4
Arizona	11.8	840	3,722.5	1.143	5.2	370.7	1,642.8
Georgia	48.7	1,268.6	3,537.4	1.097	20.6	537.3	1,498.3
Mississippi	49.2	467.1	0	1.000	19.0	180.3	0
North Carolina	56.1	405.5	0	1.088	23.6	170.3	0
Nevada	12.0	1,919	5,169.5	1.111	5.1	823.2	2,217.5
New Mexico	25.9	738.9	4,815.4	1.000	10.0	285.3	1,859.2
Oklahoma	27.9	717.5	4,311.3	1.05	11.3	290.9	1,747.8
South Carolina	53.4	821.5	3,902.6	1.109	22.9	351.8	1,671.0
Tennessee	0	1,464.4	3,470.1	1.052	0	594.8	1,409.5
Texas	31.7	695.7	4,393.8	1.139	13.9	305.9	1,932.3

(a) To convert km^2 to mi^2, multiply by 0.386102. Population density is reported in WebTRAGIS in mi^2; this table provides the conversion to km^2.

Table D-10 Columbia Generating Station Truck Route Population Density by State

State	Rural Density/mi^2	Suburban Density/mi^2	Urban Density/mi^2	Population Correction Factor	Corrected Rural Population Density/km$^{2(a)}$	Corrected Suburban Population Density/km$^{2(a)}$	Corrected Urban Population Density/km$^{2(a)}$
Idaho	32.5	529.1	0	1.143	14.3	233.5	0
Nevada	4.3	516.3	0	1.111	1.8	221.5	0
Oregon	26.8	763.0	3,446.4	1.067	11.0	314.3	1,419.8
Washington	12.8	1,631.9	3,356.3	1.111	5.5	700.0	1,439.7

(a) To convert km^2 to mi^2, multiply by 0.386102. Population density is reported in WebTRAGIS in mi^2; this table provides the conversion to km^2.

Table D-11 Dresden Nuclear Power Station Truck Route Population Density by State

State	Rural Density/mi^2	Suburban Density/mi^2	Urban Density/mi^2	Population Correction Factor	Corrected Rural Population Density/km$^{2(a)}$	Corrected Suburban Population Density/km$^{2(a)}$	Corrected Urban Population Density/km$^{2(a)}$
Arizona	10.6	474.3	0	1.143	4.7	209.3	0.0
Illinois	37.0	514.8	3,948.9	0.988	14.1	196.4	1,506.4
Iowa	53.8	658.9	4,840.8	1.050	21.8	267.1	1,962.5
Nebraska	14.4	941.5	3,732.2	1.000	5.6	363.5	1,441.0
Nevada	6.9	1,871.4	4,028.9	1.111	3.0	802.8	1,728.2
Utah	25.1	947.1	5,948.2	1.167	11.3	426.7	2,680.1
Wyoming	24.6	735.1	3,608.2	1.000	9.5	283.8	1,393.1

(a) To convert km^2 to mi^2, multiply by 0.386102. Population density is reported in WebTRAGIS in mi^2; this table provides the conversion to km^2.

Table D-12 Enrico Fermi Nuclear Generating Station Truck Route Density by State

State	Rural Density/mi^2	Suburban Density/mi^2	Urban Density/mi^2	Population Correction Factor	Corrected Rural Population Density/km$^{2(a)}$	Corrected Suburban Population Density/km$^{2(a)}$	Corrected Urban Population Density/km$^{2(a)}$
Arizona	10.6	474.3	0	1.143	4.7	209.3	0.0
Illinois	45.4	892.8	3,663.6	0.988	17.3	340.6	1,397.5
Indiana	59.7	824.7	3,868.9	1.044	24.1	332.4	1,559.5
Iowa	53.8	658.9	4,840.8	1.050	21.8	267.1	1,962.5
Michigan	51.7	751.7	0	1.000	20.0	290.2	0.0
Nebraska	14.4	941.5	3,732.2	1.000	5.6	363.5	1,441.0
Nevada	6.9	1,871.4	4,028.9	1.111	3.0	802.8	1,728.2
Ohio	51	1,457.5	4,112.7	1.010	19.9	568.4	1,603.8
Utah	25.1	947.1	5,948.2	1.167	11.3	426.7	2,680.1
Wyoming	24.6	735.1	3,608.2	1.000	9.5	283.8	1,393.1

(a) To convert km^2 to mi^2, multiply by 0.386102. Population density is reported in WebTRAGIS in mi^2; this table provides the conversion to km^2.

Table D-13 Millstone Power Station Truck Route Population Density by State

State	Rural Density/mi²	Suburban Density/mi²	Urban Density/mi²	Population Correction Factor	Corrected Rural Population Density/km²(a)	Corrected Suburban Population Density/km²(a)	Corrected Urban Population Density/km²(a)
Arizona	10.6	474.3	0	1.143	4.7	209.3	0.0
Connecticut	106.3	1,809.5	5,466.1	1	41.0	698.6	2,110.5
Illinois	45.4	892.8	3,663.6	0.988	17.3	340.6	1,397.5
Indiana	59.7	824.7	3,868.9	1.044	24.1	332.4	1,559.5
Iowa	57.4	631.8	3,702.9	1.050	23.3	256.1	1,501.2
Nebraska	14.4	941.5	3,732.2	1	5.60	363.5	1,441.0
Nevada	6.9	1,871.4	4,028.9	1.111	3.00	802.8	1,728.2
New Jersey	70.1	1,288.4	4,395.2	1.010	27.3	502.4	1,714.0
New York	31.8	2,311.4	4,824.3	1.007	12.4	898.7	1,875.7
Ohio	61.0	697.1	3,520	1.010	23.8	271.8	1,372.7
Pennsylvania	44.0	417.9	5,252.4	1.009	17.10	162.8	2,046.2
Utah	25.1	947.1	5,948.2	1.167	11.3	426.7	2,680.1
Wyoming	24.6	735.1	3,608.2	1	9.50	283.8	1,393.1

(a) To convert km² to mi², multiply by 0.386102. Population density is reported in WebTRAGIS in mi²; this table provides the conversion to km².

Table D-14 Turkey Point Nuclear Generating Station Truck Route Population Density by State

State	Rural Density/mi²	Suburban Density/mi²	Urban Density/mi²	Population Correction Factor	Corrected Rural Population Density/km²[(a)]	Corrected Suburban Population Density/km²[(a)]	Corrected Urban Population Density/km²[(a)]
Alabama	44.1	1,140	0	1.029	17.5	452.9	0.0
Arizona	11.8	840	3,722.5	1.143	5.2	370.7	1,642.8
Florida	39.9	1,178.5	4,628.6	1.127	17.4	512.8	2,014.1
Louisiana	44.4	1,126.9	5,423	1.03	17.7	448.1	2,156.6
Mississippi	39.3	658	3,905.8	1	15.2	254.1	1,508.0
Nevada	12.0	1,919	5,169.5	1.111	5.1	823.2	2,217.5
New Mexico	25.9	738.9	4,815.4	1	10.0	285.3	1,859.2
Texas	41.7	1,036.5	5,478.4	1.139	18.3	455.8	2,409.2

(a) To convert km² to mi², multiply by 0.386102. Population density is reported in WebTRAGIS in mi²; this table provides the conversion to km².

D.5 Daily Traffic Count, Truck Speeds, and Accident Rates

The NRC staff evaluated potential data sources for daily traffic counts and commercial freight transport speeds in order to apply the most current values in the transportation analysis. The most appropriate data sources that are publicly and readily available for each State included in the transportation evaluation are the interstate highway (Table D3 and Table D5 in Weiner et al. [2013-TN3390]) for daily traffic count and the "State Speed Limit Chart" provided on a National Motorists Association website for transport speed (NMA 2023-TN8064). These are provided in Table D-15. Additionally, truck accident, fatality, and injury rates are provided in Table D-16 based on website information from FHWA (2020-TN8103) and FMCSA (2022-TN8075).

Table D-15 Daily Traffic Count and Truck Speed by State

State and Route Segment	Average Traffic Count (vehicles/h)[(a)]	Transport vehicle speed (miles/h)[(b)]	Transport vehicle speed (km/h)[(b)]
AL-RURAL	1,161	70	113
AL-SUBURBAN	2,138	70	113
AL-URBAN	3,784	65	105
AR-RURAL	897	70	113
AR-SUBURBAN	1,498	70	113
AR-URBAN	3,003	65	105
AZ-RURAL	825	75	121
AZ-SUBURBAN	2,144	75	121
AZ-URBAN	4,208	65	105
CA-RURAL	1,924	55	88
CA-SUBURBAN	4,509	55	88
CA-URBAN	7,914	55	88
CO-RURAL	1,248	75	121
CO-SUBURBAN	2,342	75	121
CO-URBAN	4,051	65	105
CT-RURAL	439	65	105
CT-SUBURBAN	726	65	105
CT-URBAN	2,129	55	88
DE-RURAL	7,187	65	105
DE-SUBURBAN	3,651	65	105
DE-URBAN	3,350	55	88
FL-RURAL	1,427	70	113
FL-SUBURBAN	2,776	70	113
FL-URBAN	5,611	65	105
GA-RURAL	1,537	70	113
GA-SUBURBAN	3,286	70	113
GA-URBAN	7,340	65	105
IA-RURAL	992	70	113
IA-SUBURBAN	1,588	70	113
IA-URBAN	2,157	55	88

Table D-15 Daily Traffic Count and Truck Speed by State (Continued)

State and Route Segment	Average Traffic Count (vehicles/h)[a]	Transport vehicle speed (mph)[b]	Transport vehicle speed (km/h)[b]
ID-RURAL	1,123	70	113
ID-SUBURBAN	2,670	70	113
ID-URBAN	5,624	65	105
IL-RURAL	1,200	70	113
IL-SUBURBAN	2,466	70	113
IL-URBAN	4,408	55	88
IN-RURAL	1,200	65	105
IN-SUBURBAN	2,466	65	105
IN-URBAN	4,408	55	88
LA-RURAL	897	75	121
LA-SUBURBAN	1,498	75	121
LA-URBAN	3,003	70	113
MI-RURAL	1,219	65	105
MI-SUBURBAN	2,309	65	105
MI-URBAN	4,648	60	97
MS-RURAL	1,427	70	113
MS-SUBURBAN	2,776	70	113
MS-URBAN	5,611	70	113
NC-RURAL	1,427	70	113
NC-SUBURBAN	2,776	70	113
NC-URBAN	5,611	70	113
NE-RURAL	833	75	121
NE-SUBURBAN	1,685	75	121
NE-URBAN	3,075	70	113
NJ-RURAL	2,609	65	105
NJ-SUBURBAN	3,322	65	105
NJ-URBAN	4,527	55	88
NM-RURAL	654	75	121
NM-SUBURBAN	1,208	75	121
NM-URBAN	3,347	65	105
NV-RURAL	1,421	80	129
NV-SUBURBAN	3,732	80	129
NV-URBAN	7,517	65	105
NY-RURAL	835	65	105
NY-SUBURBAN	1,818	65	105
NY-URBAN	4,002	55	88
OH-RURAL	1,824	70	113
OH-SUBURBAN	2,655	70	113
OH-URBAN	4,241	65	105
OK-RURAL	1,175	75	121

Table D-15 Daily Traffic Count and Truck Speed by State (Continued)

State and Route Segment	Average Traffic Count (vehicles/h)[(a)]	Transport vehicle speed (mph)[(b)]	Transport vehicle speed (km/h)[(b)]
OK-SUBURBAN	1,786	75	121
OK-URBAN	2,778	70	113
OR-RURAL	1,123	65	105
OR-SUBURBAN	2,670	65	105
OR-URBAN	5,624	55	88
PA-RURAL	2,056	70	113
PA-SUBURBAN	3,655	70	113
PA-URBAN	5,748	70	113
SC-RURAL	1,427	70	113
SC-SUBURBAN	2,776	70	113
SC-URBAN	5,611	60	97
TN-RURAL	1,570	70	113
TN-SUBURBAN	2,735	70	113
TN-URBAN	4,121	65	105
TX-RURAL	897	75	121
TX-SUBURBAN	1,498	75	121
TX-URBAN	3,003	75	121
UT-RURAL	731	75	121
UT-SUBURBAN	1,958	75	121
UT-URBAN	3,940	65	105
WA-RURAL	1,123	60	97
WA-SUBURBAN	2,670	60	97
WA-URBAN	5,624	60	97
WY-RURAL	795	75	121
WY-SUBURBAN	1,956	75	121
WY-URBAN	3,708	65	105

Column one entries in this table are in the format State abbreviation-Route area segment type.
(a) Values from Weiner et al. (2013-TN3390) Tables D3 and D5 for interstate highways.
(b) Values from National Motorist Association's State Speed Limit Chart (NMA 2023-TN8064).

Table D-16 Truck Accident, Fatality, and Injury Rates

State	Rural truck miles[a] × 10^6	Urban truck miles[a] × 10^6	No. of total crashes[b] in 2021	Accidents/km	No. of fatalities[b]	Fatalities/km	No. of injuries[b]	Injuries/km
Alabama	3,532	2,527	4,483	4.60E-07	152	1.56E-08	1,677	1.72E-07
Arizona	3,217	4,349	2,567	2.11E-07	127	1.04E-08	484	3.98E-08
Arkansas	3,131	1,127	2,914	4.25E-07	101	1.47E-08	1,183	1.73E-07
California	9,027	24,199	14,096	2.64E-07	449	8.40E-09	6,679	1.25E-07
Colorado	1,700	2,319	1,888	2.92E-07	97	1.50E-08	563	8.71E-08
Connecticut	235	2,126	1,520	4.00E-07	25	6.58E-09	555	1.46E-07
Delaware	206	481	598	5.41E-07	6	5.43E-09	297	2.69E-07
Florida	6,132	14,345	9,018	2.74E-07	321	9.74E-09	4,138	1.26E-07
Georgia	4,884	10,177	5,857	2.42E-07	217	8.95E-09	2,552	1.05E-07
Idaho	1,673	421	655	1.94E-07	39	1.16E-08	360	1.07E-07
Illinois	5,178	5,239	6,634	3.96E-07	155	9.25E-09	3,238	1.93E-07
Indiana	5,380	3,079	5,647	4.15E-07	152	1.12E-08	1,678	1.23E-07
Iowa	2,968	807	2,147	3.53E-07	67	1.10E-08	763	1.26E-07
Kansas	2,767	1,048	1,774	2.89E-07	86	1.40E-08	501	8.16E-08
Kentucky	3,785	1,875	3,112	3.42E-07	107	1.18E-08	1,308	1.44E-07
Louisiana	2,049	3,117	3,850	4.63E-07	127	1.53E-08	3,045	3.66E-07
Massachusetts	261	6,172	1,782	1.72E-07	21	2.03E-09	711	6.87E-08
Michigan	2,649	3,301	5,309	5.55E-07	95	9.92E-09	1,401	1.46E-07
Mississippi	3,354	884	1,852	2.72E-07	68	9.97E-09	943	1.38E-07
Missouri	5,595	3,141	5,400	3.84E-07	144	1.02E-08	2,075	1.48E-07
Nebraska	2,124	808	605	1.28E-07	31	6.57E-09	234	4.96E-08
Nevada	1,115	1,234	690	1.83E-07	43	1.14E-08	340	9.00E-08
New Jersey	412	6,582	4,185	3.72E-07	55	4.89E-09	2,282	2.03E-07
New Mexico	3,354	1,556	876	1.11E-07	63	7.97E-09	328	4.15E-08
New York	3,091	6,847	7,459	4.66E-07	108	6.75E-09	4,684	2.93E-07
North Carolina	4,312	5,420	6,617	4.23E-07	147	9.39E-09	4,256	2.72E-07
Ohio	4,615	6,474	5,504	3.08E-07	184	1.03E-08	2,374	1.33E-07

Table D-16 Truck Accident, Fatality, and Injury Rates (Continued)

State	Rural truck miles[a] × 10^6	Urban truck miles[a] × 10^6	No. of total crashes[b] in 2021	Accidents/km	No. of fatalities[b]	Fatalities/km	No. of injuries[b]	Injuries/km
Oklahoma	4,821	2,916	3,318	2.67E-07	121	9.72E-09	1,251	1.00E-07
Oregon	2,760	1,693	1,653	2.31E-07	67	9.35E-09	467	6.52E-08
Pennsylvania	4,883	2,674	7,098	5.84E-07	161	1.32E-08	2,862	2.35E-07
South Carolina	3,073	3,550	3,176	2.98E-07	116	1.09E-08	1,867	1.75E-07
Tennessee	3,319	3,909	4,555	3.92E-07	191	1.64E-08	1,700	1.46E-07
Texas	13,906	17,001	20,534	4.13E-07	798	1.60E-08	10,829	2.18E-07
Utah	2,572	3,908	1,018	9.76E-08	51	4.89E-09	405	3.88E-08
Virginia	3,014	2,831	4,274	4.54E-07	99	1.05E-08	1,623	1.73E-07
Washington	1,937	3,511	2,170	2.48E-07	74	8.44E-09	410	4.68E-08
Wyoming	1,492	261	1,002	3.55E-07	16	5.67E-09	232	8.23E-08

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
(a) FHWA 2020-TN8103.
(b) FMCSA 2022-TN8075.

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APPENDIX E

TRANSPORTATION EVALUATION RESULTS

Table E-1 through Table E-7 provide the results for each U.S. Nuclear Regulatory Commission-Radioactive Material Transport (NRC-RADTRAN) calculation for the single shipment impacts followed by applying those values to determine the total annual normal condition and accident radiological and nonradiological transportation impacts for spent and unirradiated accident tolerant fuel (ATF). Table E-9 and Table E-12 provide the results of the NRC-RADTRAN sensitivity calculation for normal condition and accident impacts for rail shipments from each site using spent pressurized water reactor (PWR) ATF and accident impacts with greater release fractions for 72 and 85 GWd/MTU burnup levels for truck shipments from Turkey Point Nuclear Generating Station to Yucca Mountain.

Table E-1 Normal Condition and Accident Radiological Impacts per Shipment

Site (Reactor Type)	Total Miles per Shipment	Crew (person-rem)	Public Onlooker (person-rem)	Public Along Route (person-rem)	Population Accident Risk (person-rem)
Framatome FFF to Turkey Point (BWR)	3,187	3.52E-02	8.34E-02	3.38E-04	N/A
Framatome FFF to Turkey Point (PWR)[a]	3,187	3.52E-02	8.34E-02	3.38E-04	N/A
Brunswick (BWR)[a]	2,475	1.37E-01	1.69E-01	9.44E-03	9.36E-08
Columbia (BWR)[a]	908	5.08E-02	7.54E-02	1.18E-03	3.42E-09
Dresden (BWR)[a]	1,843	1.00E-01	1.05E-01	3.84E-03	3.69E-08
Fermi (BWR)[a]	2,131	1.18E-01	1.16E-01	5.95E-03	6.04E-08
Millstone (BWR)	2,770	1.56E-01	1.80E-01	1.00E-02	1.56E-07
Turkey Point (BWR)	2,642	1.46E-01	1.55E-01	1.06E-02	1.93E-07
Brunswick (PWR)	2,475	1.37E-01	1.69E-01	9.44E-03	3.19E-07
Columbia (PWR)	908	5.08E-02	7.54E-02	1.18E-03	1.16E-08
Dresden (PWR)	1,843	1.00E-01	1.05E-01	3.88E-03	1.26E-07
Fermi (PWR)	2,131	1.18E-01	1.16E-01	5.95E-03	2.06E-07
Millstone (PWR)[a]	2,770	1.56E-01	1.80E-01	1.00E-02	5.30E-07
Turkey Point (PWR)[a]	2,642	1.46E-01	1.55E-01	1.06E-02	6.57E-07

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Framatome FFF = Framatome Inc. Fuel Fabrication Facility, Turkey Point = Turkey Point Nuclear Generating Station, BWR = boiling water reactor, Brunswick = Brunswick Nuclear Generating Station, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, N/A = not applicable; PWR= pressurized water reactor.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-2 Total Annual Radiological impacts for Normal Conditions and Accidents for One-Third-Core Reload

Site (Reactor Type)	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Total Public Dose (person-rem)	Total Population Accident Risk (person-rem)
Framatome FFF to Turkey Point (BWR)	4	5.07E-02	2.72E-01	1.10E-03	2.73E-01	N/A
Framatome FFF to Turkey Point (PWR)[(a)]	3	3.80E-02	2.04E-01	8.25E-04	2.05E-01	N/A
Brunswick (BWR)[(a)]	52	2.56E+00	7.14E+00	4.00E-01	7.54E+00	4.87E-06
Columbia (BWR)[(a)]	52	9.51E-01	3.19E+00	5.01E-02	3.24E+00	1.78E-07
Dresden (BWR)[(a)]	52	1.87E+00	4.46E+00	1.63E-01	4.62E+00	1.92E-06
Fermi (BWR)[(a)]	52	2.21E+00	4.92E+00	2.52E-01	5.17E+00	3.14E-06
Millstone (BWR)	52	2.92E+00	7.61E+00	4.25E-01	8.04E+00	8.11E-06
Turkey Point (BWR)	52	2.73E+00	6.56E+00	4.49E-01	7.01E+00	1.00E-05
Brunswick (PWR)	30	1.48E+00	4.12E+00	2.31E-01	4.35E+00	9.57E-06
Columbia (PWR)	30	5.49E-01	1.84E+00	2.89E-02	1.87E+00	3.48E-07
Dresden (PWR)	30	1.08E+00	2.57E+00	9.38E-02	2.67E+00	3.78E-06
Fermi (PWR)	30	1.27E+00	2.84E+00	1.45E-01	2.99E+00	6.18E-06
Millstone (PWR)[(a)]	30	1.68E+00	4.39E+00	2.45E-01	4.64E+00	1.59E-05
Turkey Point (PWR)[(a)]	30	1.58E+00	3.78E+00	2.59E-01	4.04E+00	1.97E-05

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Framatome FFF = Framatome Inc. Fuel Fabrication Facility, Turkey Point = Turkey Point Nuclear Generating Station, BWR = boiling water reactor, Brunswick = Brunswick Nuclear Generating Station, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, PWR= pressurized water reactor, N/A = not applicable.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-3 Total Annual Radiological impacts for Normal Conditions and Accidents for Half-Core Reload

Site (Reactor Type)	No. of Normalized Annual Shipments	Worker Dose (person-rem)	Public Onlooker Dose (person-rem)	Public Along Route Dose (person-rem)	Total Public Dose (person-rem)	Total Population Accident Risk (person-rem)
Framatome FFF to Turkey Point (BWR)	6	7.60E-02	4.07E-01	1.65E-03	4.09E-01	N/A
Framatome FFF to Turkey Point (PWR)[(a)]	5	6.34E-02	3.40E-01	1.37E-03	3.41E-01	N/A
Brunswick (BWR)[(a)]	78	3.85E+00	1.07E+01	5.99E-01	1.13E+01	7.30E-06
Columbia (BWR)[(a)]	78	1.43E+00	4.79E+00	7.52E-02	4.86E+00	2.67E-07
Dresden (BWR)[(a)]	78	2.81E+00	6.69E+00	2.44E-01	6.94E+00	2.88E-06
Fermi (BWR)[(a)]	78	3.31E+00	7.38E+00	3.78E-01	7.76E+00	4.71E-06
Millstone (BWR)	78	4.38E+00	1.14E+01	6.37E-01	1.21E+01	1.22E-05
Turkey Point (BWR)	78	4.10E+00	9.83E+00	6.73E-01	1.05E+01	1.51E-05
Brunswick (PWR)	45	2.22E+00	6.18E+00	3.46E-01	6.53E+00	1.44E-05
Columbia (PWR)	45	8.23E-01	2.76E+00	4.34E-02	2.81E+00	5.22E-07
Dresden (PWR)	45	1.62E+00	3.86E+00	1.41E-01	4.00E+00	5.67E-06
Fermi (PWR)	45	1.91E+00	4.26E+00	2.18E-01	4.48E+00	9.27E-06
Millstone (PWR)[(a)]	45	2.53E+00	6.59E+00	3.68E-01	6.95E+00	2.39E-05
Turkey Point (PWR)[(a)]	45	2.37E+00	5.67E+00	3.88E-01	6.06E+00	2.96E-05

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02. Framatome FFF = Framatome Inc. Fuel Fabrication Facility, Turkey Point = Turkey Point Nuclear Generating Station, BWR = boiling water reactor, Brunswick = Brunswick Nuclear Generating Station, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, N/A = not applicable; PWR= pressurized water reactor.

(a) Denotes the reactor type at the site location under the current NRC license.

Table E-4 Nonradiological Accident Fatalities and Injury Rates

Site	One-Way Shipping Distance (miles)	One-Way Shipping Distance (km)	Accidents per Trip	Fatalities per Trip	Injuries per Trip
Brunswick (BWR)[(a)]	2,475	3,982	1.11E-03	4.46E-05	4.62E-04
Columbia (BWR)[(a)]	908	1,461	2.99E-04	1.53E-05	1.16E-04
Dresden (BWR)[(a)]	1,843	2,965	6.92E-04	2.21E-05	2.39E-04
Fermi (BWR)[(a)]	2,131	3,428	8.75E-04	2.70E-05	3.02E-04
Millstone (BWR)	2,770	4,457	1.35E-03	3.78E-05	5.07E-04
Turkey Point (BWR)	2,642	4,251	1.22E-03	4.96E-05	5.96E-04
Brunswick (PWR)	2,475	3,982	1.11E-03	4.46E-05	4.62E-04
Columbia (PWR)	908	1,461	2.99E-04	1.53E-05	1.16E-04
Dresden (PWR)	1,843	2,965	6.92E-04	2.21E-05	2.39E-04
Fermi (PWR)	2,131	3,428	8.75E-04	2.70E-05	3.02E-04
Millstone (PWR)[(a)]	2,770	4,457	1.35E-03	3.78E-05	5.07E-04
Turkey Point (PWR)[(a)]	2,642	4,251	1.22E-03	4.96E-05	5.96E-04

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station, BWR = boiling water reactor, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, Turkey Point = Turkey Point Nuclear Generating Station, PWR= pressurized water reactor.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-5 Spent Fuel Nonradiological Impacts for One-Third-Core Reload

Site (Reactor Type)	No. of Normalized Annual Truck Shipments	One-Way Shipping Distance (miles)	One-Way Shipping Distance (km)	Annual Round Trip Accidents	Annual Round Trip Fatalities	Annual Round Trip Injuries
Brunswick (BWR)[(a)]	52	2,475	3,982	1.15E-01	4.64E-03	4.80E-02
Columbia (BWR)[(a)]	52	908	1,461	3.11E-02	1.59E-03	1.21E-02
Dresden (BWR)[(a)]	52	1,843	2,965	7.20E-02	2.30E-03	2.49E-02
Fermi (BWR)[(a)]	52	2,131	3,428	9.10E-02	2.81E-03	3.14E-02
Millstone (BWR)	52	2,770	4,457	1.40E-01	3.93E-03	5.27E-02
Turkey Point (BWR)	52	2,642	4,251	1.27E-01	5.16E-03	6.20E-02
Brunswick (PWR)	30	2,475	3,982	6.66E-02	2.68E-03	2.77E-02
Columbia (PWR)	30	908	1,461	1.79E-02	9.18E-04	6.96E-03
Dresden (PWR)	30	1,843	2,965	4.15E-02	1.33E-03	1.43E-02
Fermi (PWR)	30	2,131	3,428	5.25E-02	1.62E-03	1.81E-02
Millstone (PWR)[(a)]	30	2,770	4,457	8.10E-02	2.27E-03	3.04E-02
Turkey Point (PWR)[(a)]	30	2,642	4,251	7.32E-02	2.98E-03	3.58E-02

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station, BWR = boiling water reactor, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, Turkey Point = Turkey Point Nuclear Generating Station, PWR= pressurized water reactor.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-6 Spent Fuel Nonradiological Impacts for Half-Core Reload

Site (Reactor Type)	No. of Normalized Annual Truck Shipments	One-Way Shipping Distance (miles)	One-Way Shipping Distance (km)	Annual Round Trip Accidents	Annual Round Trip Fatalities	Annual Round Trip Injuries
Brunswick (BWR)[(a)]	78	2,475	3,982	1.73E-01	2.21E-01	3.68E+00
Columbia (BWR)[(a)]	78	908	1,461	4.66E-02	2.78E-02	3.39E-01
Dresden (BWR)[(a)]	78	1,843	2,965	1.08E-01	8.15E-02	1.42E+00
Fermi (BWR)[(a)]	78	2,131	3,428	1.37E-01	1.15E-01	2.07E+00
Millstone (BWR)	78	2,770	4,457	1.22E-01	2.09E-01	4.52E+00
Turkey Point (BWR)	78	2,642	4,251	1.90E-01	2.62E-01	5.07E+00
Brunswick (PWR)	45	2,475	3,982	9.99E-02	2.21E-01	3.68E+00
Columbia (PWR)	45	908	1,461	2.69E-02	2.78E-02	3.39E-01
Dresden (PWR)	45	1,843	2,965	6.23E-02	8.15E-02	1.42E+00
Fermi (PWR)	45	2,131	3,428	7.88E-02	1.15E-01	2.07E+00
Millstone (PWR)[(a)]	45	2,770	4,457	2.11E-01	2.09E-01	4.52E+00
Turkey Point (PWR)[(a)]	45	2,642	4,251	1.10E-01	2.62E-01	5.07E+00

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station, BWR = boiling water reactor, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, Turkey Point = Turkey Point Nuclear Generating Station, PWR= pressurized water reactor.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-7 Unirradiated Fuel Nonradiological Impacts for One-Third-Core Reload

Site	Normalized Annual Truck Shipments	One-Way Shipping Distance (miles)	One-Way Shipping Distance (km)	Accidents/ Trip	Fatalities/ Trip	Injuries/ Trip	Annual Accidents	Annual Fatalities	Annual Injuries
Framatome FFF to Turkey Point (BWR)	4	3,187	5,128	1.38E-03	4.64E-05	5.34E-04	1.10E-02	3.71E-04	4.27E-03
Framatome FFF to Turkey Point (PWR)[(a)]	3	3,187	5,128	1.38E-03	4.64E-05	5.34E-04	8.28E-03	2.78E-04	3.20E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Framatome FFF = Framatome Inc. Fuel Fabrication Facility, Turkey Point = Turkey Point Nuclear Generating Station, BWR = boiling water reactor, PWR= pressurized water reactor.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-8 Unirradiated Fuel Nonradiological Impacts for Half-Core Reload

Site	Normalized Annual Truck Shipments	One-Way Shipping Distance (miles)	One-Way Shipping Distance (km)	Accidents/ Trip	Fatalities/ Trip	Injuries/ Trip	Annual Accidents	Annual Fatalities	Annual Injuries
Framatome FFF to Turkey Point (BWR)	6	3,187	5,128	1.38E-03	4.64E-05	5.34E-04	1.66E-02	5.57E-04	6.41E-03
Framatome FFF to Turkey Point (PWR)[(a)]	5	3,187	5,128	1.38E-03	4.64E-05	5.34E-04	1.38E-02	4.64E-04	5.34E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Framatome FFF = Framatome Inc. Fuel Fabrication Facility, Turkey Point = Turkey Point Nuclear Generating Station, BWR = boiling water reactor, PWR= pressurized water reactor.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-9 Spent Accident Tolerant Fuel Rail Transportation Impacts for One-Third-Core Reload

Site	No. of Normalized Annual Shipments	One-Way Shipping Distance (miles)	Crew (person-rem/ shipment)	Public Onlooker (person-rem/ shipment)	Public Along Route (person-rem/ shipment)	Population Risk (person-rem/ shipment)	Total Annual Crew Dose (person-rem)	Total Annual Public Onlooker Dose (person-rem)	Total Annual Public Along Route Dose (person-rem)	Total Annual Public Dose (person-rem)	Total Annual Accidental Population Risk (person-rem)
Brunswick (PWR)	1.25	3,009	1.20E-03	6.88E-04	1.64E-02	2.51E-11	2.16E-02	8.60E-04	2.05E-02	2.14E-02	7.53E-10
Columbia (PWR)	1.25	1,218	4.86E-04	2.12E-04	4.54E-03	7.24E-12	1.10E-02	2.65E-04	5.68E-03	5.94E-03	2.17E-10
Dresden (PWR)	1.25	1,933	7.72E-04	3.71E-04	7.49E-03	1.12E-11	1.52E-02	4.64E-04	9.36E-03	9.83E-03	3.36E-10
Fermi (PWR)	1.25	2,334	9.32E-04	5.10E-04	1.24E-02	2.12E-11	1.77E-02	6.38E-04	1.55E-02	1.61E-02	6.36E-10
Millstone (PWR)[a]	1.25	2,975	1.19E-03	7.17E-04	1.90E-02	3.11E-11	2.14E-02	8.96E-04	2.38E-02	2.46E-02	9.33E-10
Turkey Point (PWR)[a]	1.25	3,311	1.32E-03	7.64E-04	2.01E-02	3.39E-11	2.34E-02	9.55E-04	2.51E-02	2.61E-02	1.02E-09

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station, PWR= pressurized water reactor, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-10 Spent Accident Tolerant Fuel Rail Transportation Impacts for Half-Core Reload

Site	No. of Normalized Annual Shipments	One-Way Shipping Distance (miles)	Crew (person-rem/ shipment)	Public Onlooker (person-rem/ shipment)	Public Along Route (person-rem/ shipment)	Population Risk (person-rem/ shipment)	Total Annual Crew Dose (person-rem)	Total Annual Public Onlooker Dose (person-rem)	Total Annual Public Along Route Dose (person-rem)	Total Annual Public Dose (person-rem)	Total Annual Accidental Population Risk (person-rem)
Brunswick (PWR)	1.73	3,009	1.20E-03	6.88E-04	1.64E-02	2.51E-11	2.99E-02	1.19E-03	2.84E-02	2.96E-02	1.04E-09
Columbia (PWR)	1.73	1,218	4.86E-04	2.12E-04	4.54E-03	7.24E-12	1.52E-02	3.67E-04	7.85E-03	8.22E-03	3.01E-10
Dresden (PWR)	1.73	1,933	7.72E-04	3.71E-04	7.49E-03	1.12E-11	2.11E-02	6.42E-04	1.30E-02	1.36E-02	4.65E-10
Fermi (PWR)	1.73	2,334	9.32E-04	5.10E-04	1.24E-02	2.12E-11	2.44E-02	8.82E-04	2.15E-02	2.23E-02	8.80E-10
Millstone (PWR)[(a)]	1.73	2,975	1.19E-03	7.17E-04	1.90E-02	3.11E-11	2.96E-02	1.24E-03	3.29E-02	3.41E-02	1.29E-09
Turkey Point (PWR)[(a)]	1.73	3,311	1.32E-03	7.64E-04	2.01E-02	3.39E-11	3.24E-02	1.32E-03	3.48E-02	3.61E-02	1.41E-09

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
Brunswick = Brunswick Nuclear Generating Station, PWR= pressurized water reactor, Columbia = Columbia Generating Station, Dresden = Dresden Generating Station, Fermi = Enrico Fermi Nuclear Generating Station, Millstone = Millstone Nuclear Power Plant, Turkey Point = Turkey Point Nuclear Generating Station.
(a) Denotes the reactor type at the site location under the current NRC license.

Table E-11 Burnup Release Fractions Sensitivity Analysis Results for One-Third-Core Reload

Reactor Type – Burnup	No. of Normalized Annual Shipments	Crew (person-rem)/shipment	Public Onlooker (person-rem)/shipment	Public Along Route (person-rem)/shipment	Population Risk (person-rem)/shipment	Total Annual Worker Dose (person-rem)	Total Annual Public Onlooker Dose (person-rem)	Total Annual Public Along Route Dose (person-rem)	Total Annual Public Dose (person-rem)	Total Annual Accidental Population Risk (person-rem)
BWR — 72 GWd/MTU	52	1.46E-01	1.55E-01	1.06E-02	3.95E-05	2.73E+00	6.56E+00	4.49E-01	7.01E+00	2.05E-03
BWR — 85 GWd/MTU	52	1.46E-01	1.55E-01	1.06E-02	7.63E-05	2.73E+00	6.56E+00	4.49E-01	7.01E+00	3.97E-03
PWR — 72 GWd/MTU	30	1.46E-01	1.55E-01	1.06E-02	4.33E-05	1.58E+00	3.78E+00	2.59E-01	4.04E+00	1.30E-03
PWR — 85 GWd/MTU	30	1.46E-01	1.55E-01	1.06E-02	8.29E-05	1.58E+00	3.78E+00	2.59E-01	4.04E+00	2.49E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
GWd/MTU= gigawatt days per metric ton of uranium, BWR = boiling water reactor, PWR= pressurized water reactor.
All sensitivity cases are spent accident tolerant fuel truck shipments from the Turkey Point Nuclear Generating Station site to Yucca Mountain.

Table E-12 Burnup Release Fractions Sensitivity Analysis Results for Half-Core Reload

Reactor Type – Burnup	No. of Normalized Annual Shipments	Crew (person-rem)/shipment	Public Onlooker (person-rem)/shipment	Public Along Route (person-rem)/shipment	Population Risk (person-rem)/shipment	Total Annual Worker Dose (person-rem)	Total Annual Public Onlooker Dose (person-rem)	Total Annual Public Along Route Dose (person-rem)	Total Annual Public Dose (person-rem)	Total Annual Accidental Population Risk (person-rem)
BWR — 72 GWd/MTU	78	1.46E-01	1.55E-01	1.06E-02	3.95E-05	4.10E+00	9.83E+00	6.73E-01	1.05E+01	3.08E-03
BWR — 85 GWd/MTU	78	1.46E-01	1.55E-01	1.06E-02	7.63E-05	4.10E+00	9.83E+00	6.73E-01	1.05E+01	5.95E-03
PWR — 72 GWd/MTU	45	1.46E-01	1.55E-01	1.06E-02	4.33E-05	2.73E+00	5.67E+00	3.88E-01	6.06E+00	1.95E-03
PWR — 85 GWd/MTU	45	1.46E-01	1.55E-01	1.06E-02	8.29E-05	2.73E+00	5.67E+00	3.88E-01	6.06E+00	3.73E-03

Scientific notation is denoted by E followed by the exponent. For example, 5.02×10^{-2} is indicated by 5.02E-02.
GWd/MTU= gigawatt days per metric ton of uranium, BWR = boiling water reactor, PWR= pressurized water reactor.
All sensitivity cases are spent accident tolerant fuel truck shipments from the Turkey Point Nuclear Generating Station site to Yucca Mountain.

APPENDIX F

COMMENTS RECEIVED ON THE ENVIRONMENTAL EVALUATION OF ACCIDENT TOLERANT FUELS WITH INCREASED ENRICHMENT OF HIGHER BURNUP LEVELS, NUREG-2266

F.1 Comments Received During the Public Comment Period

On August 31, 2023, the U.S. Nuclear Regulatory Commission (NRC) issued and distributed Draft NUREG-2266, “Environmental Evaluation of Accident Tolerant Fuels with Increased Enrichment of Higher Burnup Levels,” to interested members of the public. In addition, the NRC issued its Notice of Availability Draft NUREG-2266 for public comment on September 1, 2023 (88 FR 60507-TN9593). The public comment period ended on October 31, 2023.

At the end of the draft NUREG-2266 (NRC 2023-TN9589) public comment period, the staff collected the comments on the draft NUREG-2266 as listed in Table F-1. Each commenter is identified by the commenter’s ID number and comment source document number in ADAMS. The staff updated or revised the information in the NUREG-2266 as appropriate and has issued this NUREG as final.

Table F-1 Individuals Providing Comments During the Comment Period

Commenter’s ID	Last Name	First Name	Affiliation	Comment Source	ADAMS Accession No.
1	Baker	Ryan	Private Citizen	Regulations.gov	ML23298A153
2	Pimentel	Frances A.	Nuclear Energy Institute	Regulations.gov	ML23305A155
3	Harper	Zachary S.	Westinghouse Electric Company	Regulations.gov	ML23305A157
4	Walker	Kalene	Private Citizen	Regulations.gov	ML23305A158

The NRC staff has organized its responses to public comments according to the following subjects of comment on the draft NUREG-2266:

- Editorial Comments
- Opposition
- Outside Scope
- Support
- Transportation
- Uranium Fuel Cycle

Note to the reader: the individual comments begin with “**Comment:**” and were entered verbatim and formatted to be consistent with the comment in the referenced documents in Table F-1. The staff’s responses to the comments follow the corresponding public comments and begin with “***Response:***.” To further aid the reader, the staff’s response are *in italics.*

F.1.1 Editorial Comments and Responses

Comment:
Comment Number: 4
Section Number: 5.0 Conclusion (Page 5-2)
Comment/Basis: Line 31-32 states, "In particular, applicants must discuss whether: ..."
Recommendation: Change the word "must" to "should" (**2-6** [Pimentel, Frances A.])

Comment:
Comment Number: 8
Section Number: B.9 Crud Release (Page B-8)
Comment/Basis: Change modern PWRs to say modern LWRs in Line 12.
Recommendation: Change modern PWRs to say modern LWRs in Line 12. (**2-10** [Pimentel, Frances A.])

Comment:
Page Number: B-1
Section Number**:** B.1
Comment**:** Line 17: Recommend that "spent fuel case accidents" should be changed to "spent fuel cask accidents" (**3-1** [Harper, Zachary S.])

Comment:
Page Number: 1-5, 1-7
Section Number**:** 1.4.2, 1.4.4
Comment: Recommend choosing one spelling for the vendor/ vender; both spellings are currently used. "Vender" is currently used on pages 1-5 and 1-7, "Vendor" is used for all other instances. (**3-8** [Harper, Zachary S.])

Comment:
Page Number: 5-2
Section Number: 5
Comment: Line 20: Change "bound" to "bounded" (**3-9** [Harper, Zachary S.])

Response: *These comments provided corrections to typographical errors and suggestions for word choice. The NRC staff reviewed the context within the document and made the suggested corrections for “bound” to “bounded,” “must” to “should,” “PWRs” to “LWRs,” and “vender” to “vendor” in the appropriate locations within the NUREG-2266. For “spent fuel* ***case*** *accidents,” since it has been used in the context of transportation, NRC’s preferred term is “spent fuel* ***package*** *accidents” and this change was made. Changes were made to the NUREG-2266 based on these comments.*

Comment:
Comment Number: 5
Section Number: 5.0 Conclusion (Page 5-2)
Comment/Basis: This section includes items that must be discussed as part of a generic evaluation. Lines 40-42 indicate the applicant discuss "the number of annual unirradiated and spent ATF shipments over the refueling cycle time being requested in the LAR application based on the expected transport package fall within the number of shipments discussed in this study.” Since the number of shipments has a lot of variability, it is suggested to add the word “generally”

before “based on the expected transport package fall within the number of shipments discussed in this study.”
Recommendation: Add the word "generally" before "based on the expected transport package fall within the number of shipments discussed in this study." The bullet in final form would be: "The number of annual unirradiated and spent ATF shipments over the refueling cycle time being requested in the LAR application generally based on the expected transport package fall within the number of shipments discussed in this study." (**2-7** [Pimentel, Frances A.])

Response: *The NRC staff acknowledges the comment but instead of adding the word “generally” as recommended, the NRC staff supplemented the text on how licensees should consider the number of shipments when using this NUREG in future licensing amendment request applications. Because of the importance of the number of shipments in assessing the significance of the impacts from incident-free, normal operation, fresh, and spent accident tolerant fuel (ATF) transportation, the NRC staff views this value as a key factor in determining that this NUREG bounds the incident-free shipments being considered in a licensing amendment requesting deployment and use of ATF with increased enrichment and higher burnup levels. Section 5 of NUREG-2266 was modified based on this comment.*

Comment:
Comment Number: 9
Section Number**:** D.4 Population Density Adjustments (Page D-9)
Comment/Basis: This section provides examples of the data sources used by WebTRAGIS when developing the population datasets. One is called Census TIGER road data. Please define "TIGER" as it is not obvious as what this refers to.
Recommendation: Define "TIGER" in context of "Census TIGER road data."
(**2-11** [Pimentel, Frances A.])

Response**:** *The staff updated the NUREG-2266 to state that “TIGER” is the acronym for “Topologically Integrated Geographic Encoding and Referencing” system and represents the U.S. Census Bureau’s geographic spatial data (USCB 2023-TN9586). TIGER data products are developed using the TIGER geospatial data as the primary source. NUREG-2266 content and the list of acronyms and abbreviations were modified based on this comment.*

Comment:
Page Number: 2-11
Section Number: 2.2.5.1
Comment: Line 39: Recommend adding a footnote to define ''environmental justice.'' This definition could also be provided in Section 2.3.1, Considerations of Environmental Justice. (**3-2** [Harper, Zachary S.])

Response: *The NRC defined “Environmental Justice” in Section 11, Glossary of the NUREG-2157 as:*

> *The fair treatment of people of all races, cultures, incomes, and educational levels with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies.*

This definition of “Environmental Justice” was added to Section 2.2.5.1 of NUREG-2266.

Comment:
Page Number: 2-11
Section Number: 2.2.5.1
Comment: Line 39: Recommend clarifying what the "climate change" impacts are limited to (e.g., greenhouse gas emissions (GHG)). GHG is discussed in Section 2.3.2 as an addition to Table S-3 of the environmental impacts assessment, but it is not clearly tied to the concept of climate change. (**3-3** [Harper, Zachary S.])

Response: *The NRC defined "Climate Change" in Section 11, Glossary of the NUREG-2157 as:*

> *Changes in the Earth's surface temperature thought to be caused by the greenhouse effect and responsible for changes in global climate patterns. The greenhouse effect is the trapping and buildup of heat in the atmosphere (troposphere) near the Earth's surface. Some of the heat flowing back toward space from the Earth's surface is absorbed by water vapor, carbon dioxide, ozone, and certain other gases in the atmosphere and then reradiated back toward the Earth's surface.*

This definition of "Climate Change" was added to Section 2.1.2 of NUREG-2266.

Comment:
Page Number: 5-2
Section Number: 5
Comment: Lines 33 through 45: Recommend that the bullet points provided in the conclusions be expanded upon to include additional information. Examples of potential improvements:
-Bullet 2 discusses the front end of the uranium fuel cycle, which is outside of the scope of the reactor owner. Recommend including a clarification on which organization should be responding/responsible for this within an ATF-related LAR.
-Bullets 3 and 4 would benefit from internal cross references to other sections and tables within The Report. For Example, Bullet 3 could reference tables in Appendix E, and Bullet 4 could reference Table 3-2 and Section D.3. (**3-6** [Harper, Zachary S.])

Response: *The NRC staff acknowledges the comment and the staff modified the bullets to address this comment and also to address other considerations by the staff. The front end of the uranium fuel cycle is well established and fully disclosed to all stakeholders. Applicants and licensees should know how their uranium fuel supply fits within the front end of the uranium fuel cycle analysis in this NUREG and so state in their application under oath and affirmation. The staff added the cross-references to other sections and tables in the appendices. Changes were made to Section 5 of NUREG-2266 based on this comment.*

Comment:
Page Number: 5-3
Section Number: 5
Comment: Line 25: The final statement of The Report's conclusions states "Additionally, if in a future licensing action, the enrichment and burnup levels are greater than 8 wt% U-235 and 80 GWd/MTU, respectively, and for the deployment and use of long-term ATF technologies, the study could provide guidance for completing the needed revised analysis." Recommend clarifying whether this statement is saying that the licensee should use this report as a basis for a revised analysis, or that the NRC will consider extending this report to cover a wider range of conditions. (**3-7** [Harper, Zachary S.])

Response: *Based on the staff revision of addressing up to 10 wt% U-235 for the uranium fuel cycle and decommissioning, this comment now only applies to the transportation of ATF and wastes where the enrichment level could not be taken above 8 wt% U-235. The conditions will be exceeding where Table S-4 of Title 10 of the Code of Federal Regulations (10 CFR) 51.52 (TN250) was determined to be bounding for the environmental effects for the transportation of fuel and wastes. In this case, 10 CFR 51.52(b) would apply. Applicants and licensees would need to provide a site-specific, full description and detailed analysis of the environmental effects of transportation of fuel and wastes to and from the reactor. The analysis would need to include values for the environmental impacts under normal conditions of transport and for the environmental risk from accidents.*

The NRC staff considers that a licensee could apply the methodology addressed in NUREG-2266 to inform a full description and detailed analysis of the impacts of transporting fuel greater than 8 wt% or 80 GWd/MTU. This involves generating radionuclide inventories and release fractions for the higher enrichment and burnup levels as addressed in Appendix A and Appendix B, site route selection using Appendix C, and NRC-RADTRAN data and input parameter values using Appendix D, including population density.

Comment:
Page Number: 3-35
Section Number: 3.8
Comment: Lines 42 through 45: It is not clear in this conclusion that a detailed site-specific transportation analysis <u>is not required</u> in the LAR application, if the LAR changes are bounded by this Report (i.e., enrichment and burnup levels). This section states "This conclusion would need to be validated in the review of an NRC licensee's LAR application..." which is unclear when compared to the executive summary. Recommend modifying the language on Page 3-35 to clarify that a site-specific transportation analysis is not required, as well as what will need to be validated as part of the NRC's review.

The executive summary states (page xvi, lines 22 through 26) "Therefore, the results of this analysis could serve as a reference in helping to address the environmental impacts of ATF licensing <u>without a detailed site-specific transportation analysis</u>..." (**3-10** [Harper, Zachary S.])

Response: *The NRC staff modified the last paragraph in Section 3.8 to read as follows:*

> *For this analysis to be applied in a future licensing action, such as for the approval of the deployment and use of ATF with increased enrichment and higher burnup levels at a licensed NPP, the application would need to confirm the licensee’s shipments are bounded by the key parameters of the transport package analyzed here. These parameters are the radionuclide inventory (see Appendix A) based on the applied enrichment and burnup levels, the number of unirradiated fuel shipments, and the number of spent fuel shipments (see Table 3-2). In that case, Table S-4 would apply. If the values associated with the contemplated shipments exceed the above-discussed values, a full description and detailed analysis of the environmental effects of transportation of fuel and wastes as required by 10 CFR 51.52(b) could be performed following the methodology of this NUREG and provided in the application.*

F.1.2 Opposition Comments and Responses

Comment: NUREG 2266 will only aggravate the serious national spent fuel waste problem. (**4-1** [Walker, Kalene])

Response: *This comment did not provide any new information related to the environmental effects of the deployment and use of ATF with increased enrichment and higher burnup levels. The NRC regulations concerning the management of spent nuclear fuel from NRC-licensed nuclear power plants provide for reasonable assurance of adequate protection of public health and safety. No changes were made to NUREG-2266 as a result of this comment.*

Comment: **Other significant issues NUREG 2266 does not address include:**

Source terms - Source terms for ATF at the higher burnups have not been analyzed and are still under developmen[t] (**4-6** [Walker, Kalene])

Response: *The NRC disagrees with this comment. NUREG-2266 did consider the potential source terms from a transportation accident as provided in Appendix A and analyzed in Section 3. If this comment is related to the accident source term associated with a nuclear reactor using ATF, that is a site-specific accident analysis. Applying such an accident analysis in this NUREG is outside the scope of this NUREG since it would not be generic due to the need to include site-specific information, such as the population distribution around an NPP site. No changes were made to NUREG-2266 based on this comment.*

Comment: NUREG 2266 does not establish a "bounding analysis" or the technical safety basis to be approved. **NRC's suggestion, through this NUREG, of broad sweeping approval of MUCH higher burnup and experimental ATF fuel would be a complete abdication of the NRC's job and responsibility.**

Because this NUREG could negatively affect many individual facilities, reactors and ISFSI's over a large geographical area over the long term (due to the forever nature of highly radioactive spent fuel waste), **NUREG 2266 requires a Programatic Environmental Impact Statement (PEIS).** (**4-9** [Walker, Kalene])

Response: *The NRC disagrees with this comment. NUREG-2266 addresses generic environmental issues that are expected at any light-water reactor (LWR) from the deployment and use of ATF with increased enrichment and higher burnup levels if industry submits the appropriate licensing amendment requests. Other environmental resource areas are plant- and site-specific, such as radiological and any physical changes, and must be assessed during the licensing action for the NRC to approve the use ATF along with any proposed increase in enrichment and higher burnup levels. Therefore, a generic or programmatic EIS is not practicable given the uncertainty in knowing all aspects of ATF deployment for a specific nuclear power plant. No changes were made to NUREG-2266 based on this comment.*

F.1.3 Comments Concerning Issues Outside Scope

Comment: Please ensure you address the reduced environmental impacts from fuel that reduces the overall risk of the reactor. By being more accident tolerant, the fuel is less likely to negatively impact the environment. The licensee should be able to adjust their actions and requirements based on the reduced risk. This could mean relaxing certain requirements, such that the net effect is the same environmental risk as before. This flexibility and consideration should be built into the rule making. (**1-1** [Baker, Ryan])

Comment: NUREG 2266 proposes to allow the generation of new, MUCH higher burnup spent fuel waste, while ignoring known and unresolved urgent problems with existing high burnup spent fuel waste and canisters. The NRC knows but ignores that existing high burnup fuel causes in-reactor hydride formation in the cladding, and on-going degradation mechanisms in storage, such as hydride reorientation, thinning and embrittlement of cladding. And the NRC continues to ignore the urgency of canister degradation concerns, including the lack of technology to identify, prevent or stop canister cracking.
The NRC has never provided a technical response to evidence of explosion risks with a breached canister. ML18269A037

The 2019 DOE Gap Analysis Report established high priority gaps. The highest priority gaps include canister corrosion, monitoring, assessment of consequence of canister failure, fuel transfer options, cladding hydrides, hydride reorientation, cladding embrittlement and fuel transfer options. https://www.osti.gov/servlets/purl/1592862

NUREG 2266's referenced **Continued Storage GEIS** document fails to address these issues. (**4-2** [Walker, Kalene])

Comment: NUREG 2266 proposes allowing experimental ATF and MUCH higher burnups.
The NRC knows that fuel with much higher burnup can suffer Fuel Fragmentation Relocation Distribution (FFRD) in a Loss of Cooling Accident (LOCA).
This could indicate that ATF is less (not more) resistant to a nuclear incident.

NUREG 2266 and supporting documents contain no technical documents showing that ATF cladding will prevent FFRD and / or cladding bursts in a LOCA.

No mitigation technology or strategy has been presented if significant numbers of cladding were to suffer FFRD and burst in the reactor. In addition, questions regarding containment and storage of damaged FFRD fuel remains unaddressed: What technology exists that could gather dispersed fine particles and small chunks of uranium pellets? How would the pulverized and dispersed FFRD fuel from large break be retrieved from the reactor? How would FFRD fuel be stored? In a spent fuel pool? In dry storage? What specific damaged fuel container systems could be used for fine highly irradiated uranium particles?
Justifications based on hope (of no LOCA occurring) is unacceptable. (**4-3** [Walker, Kalene])

Comment: Industry claims ATF will minimize cladding degradation issues.
But NRC's own <u>Interim Staff Guidance, ATF ISG 2020-01, Appendix C</u>, (ML19343A121) outlines a lengthy list of potential new damage mechanisms with chromium cladding.
For example, cracking and delamination contributing to nucleation sites could have the potential to cause hot spots and localized corrosion.
Also, "As described in Section 6.2.2 of the PIRT report, chromium coating may also impact the fuel rod ballooning characteristics under accident conditions. While no regulatory limits are currently defined to limit the extent of ballooning or the size of the rupture opening, concerns related to fuel fragmentation, relocation, and dispersal may warrant future SAFDLs for fuel rod burnup extensions beyond rod-average values of **62 gigawatt days per metric ton unit."**
NUREG 2266 suggests approving burn levels up to **80 GWd/MTU**.

DOE's High Level Gap Analysis for Accident Tolerant and Advanced Fuels for Storage and Transportation / Spent Fuel and Waste Disposition document states
"The lack of data with respect to potential storage and transportation degradation mechanisms for ATF/AF, especially for the expected higher burnups, higher temperatures and higher internal rod pressures, require a testing program... to ensure that the NRC requirement for preventing

gross rupture is met."
"Based on the current knowledge of ATF cladding and fuel designs, attention should focus on damaged spent fuel particulate size and quantity; cladding coating robustness and potential corrosion and hydride potential in areas of damaged cladding coatings..."
The DOE's report lists gaps in knowledge and data which include: Consequence of Canister Failure, Fuel Fragmentation, Fuel Restructuring/Swelling, Fuel Oxidation, Creep, Embrittlement, Thermal Cycling, Hydrogen effects: Embrittlement and Reorientation, Delayed Hydride Cracking, Oxidation, and Wet Corrosion, Thermal Profiles, Stress Profiles, Drying Issues, Fuel Transfer Options. (https://www.osti.gov/servlets/purl/1813674) (**4-4** [Walker, Kalene])

Comment: Chromium coated zirconium is in experimental stages and should not be credited for stopping hydride formation or cladding degradation.
The NRC must not ignore evidence showing that ATF fuels are insufficiently analyzed. (**4-5** [Walker, Kalene])

Comment: Power Uprates - Industry plans for half the existing reactors to increase power uprates to double the capacity of existing aged reactors. slide 7 ML23242A078.
The NUREG fails to mention power uprates, or the potential "major and costly modifications, such as replacement of main turbines...and analyses which span many technical disciplines and may be complex..." that must be reviewed by the NRC before license amendments would be permitted. "Components such as pipes, valves, pumps, heat exchangers, electrical transformers and generators must be able to accommodate the conditions that would exist at the higher power levels." https://www.nrc.gov/reactors/operating/licensing/power-uprates/about-power.html (**4-7** [Walker, Kalene])

Comment: Where is the NRC's analysis of thermal shock to an aged, uprated embrittled reactor?

The NUREG Summary states, "To support efficient and effective licensing reviews of new accident tolerant fuels (ATFs) and **to reduce the need for a complex site-specific environmental review** for each ATF license amendment request, this study evaluated the likely impacts of near-term ATF technologies with increased enrichment and higher burnup levels on the uranium fuel cycle, transportation of fuel and waste, and decommissioning of light-water reactors (LWRs) (i.e., **a bounding analysis**)." (**4-8** [Walker, Kalene])

Response: *The comments are out of scope of NUREG-2266 because they identify plant-specific safety issues related to the reduction of reactor risks from deployment of ATF, adequacy of fuel qualification and testing, core fuel performance under accident situations, cladding degradation, power uprates, thermal shock analysis, mitigation of damaged fuel, and spent fuel storage canister and transportation package failure mechanisms. All such safety issues are areas of review the staff needs to consider related to siting, fuel design and qualification, and transportation package certification before or when a licensee submits a license amendment request (LAR) for the deployment and use of ATFs with increased enrichment and higher burnup levels.*

Industry has publicly indicated that the requests to deploy and use ATF may be paired with or followed by power uprate license amendment requests. However, for the purpose of this assessment, the NRC staff is treating them separately. Environmental impacts from such *actions as a power uprate would be addressed during the staff's environmental review of the LAR requesting the power uprate. This is true for LARs that seek both a power uprate and the adoption and deployment of ATF with increased enrichment and higher burnup levels*

Apart from its review under NEPA for which this NUREG may be used, the staff would carry out its responsibilities under the Atomic Energy Act, as amended, 10 CFR regulations, and guidance for the technical safety basis to reach a separate safety determination on the deployment and use of ATF. Fuel vendors and licensed nuclear power plant operators would be required to meet applicable regulatory requirements before any deployment of ATF.

NUREG-2266 addresses environmental issues that are expected from the deployment and use of ATF with increased enrichment and higher burnup levels if industry submits the appropriate licensing amendment requests. These issues should be common to all LWRs. NUREG-2266 does not propose the approval of any extended burnup operation. This NUREG takes into account NRC regulations concerning the management of spent nuclear fuel from NRC-licensed nuclear power plants, and also accounts for the material conditions of the spent nuclear fuel. Compliance with these regulations provides reasonable assurance of adequate protection of public health and safety.

The storage conditions that a spent fuel assembly would be subject to once placed into the spent fuel pool and later in a dry storage system would be significantly less severe than the in-reactor thermal and physical stress conditions. NRC has ongoing research programs and interactions with fuel vendors, licensees, and the DOE to address the long-term conditions of higher burnup levels above 62 GWd/MTU to provide adequate protection of public health and safety prior to approving them. As described in DOE's "High Level Gap Analysis for Accident Tolerant and Advanced Fuels for Storage and Transportation / Spent Fuel and Waste Disposition" (Honnold et al. 2021-TN10172), the DOE in conjunction with the industry is currently conducting additional research to explore long-term conditions of fuels with higher burnup levels above 62 GWd/MTU, with meetings scheduled between the NRC and DOE to discuss this as research progresses. The NRC may publish the results as data become available in the future.

The above comment that the NRC has not provided a technical response to evidence of explosion risks with a breached canister referencing ML18269A037 concerns a comment submitted on the draft of NUREG-2224, "Dry Storage and Transportation of High Burnup Spent Nuclear Fuel, Draft Report for Comment" (NRC 2020-TN9594). The NRC staff responded to the comments on NUREG-2224 in a separate NRC staff document under ADAMS Accession Number ML20120A444. Responses 3.1.1, 3.1.2, and 3.1.5 addressed several comments similar to the comments received on this NUREG.

Licensees (fuel vendors or LWR licensees) will need to produce and justify specified acceptable fuel design limits (SAFDLs) to prevent ATF failures under normal operation conditions and anticipated operational occurrences (AOOs) to meet NRC safety requirements. Additionally, licensees would need to provide analyses, such as those described in Regulatory Guide 1.183 (NRC 2023-TN9587), to estimate fuel failures during in-reactor accident conditions. The NUREG-2224, Interim Staff Guidance, ATF ISG 2020-01, Appendix C, (NRC 2020-TN9603), and DOE's high level gap document were published after the Continued Storage GEIS. These documents demonstrate the continued efforts of the NRC to protect the public health and safety with regard to high burnup spent nuclear fuel, under design-basis in-reactor, storage, or transportation conditions.

Therefore, these comments were outside the scope for this NUREG. No changes were made to NUREG-2266 as a result of these comments.

F.1.4 Comments in Support of NUREG-2266 and Responses

Comment: This work will support efficient and effective licensing reviews of ATF and reduce the need for a complex site-specific environmental review for each ATF license amendment request (LAR). (**2-1** [Pimentel, Frances A.])

Response: *The staff acknowledges the comment. Because the comment did not provide new information, no changes were made to NUREG-2266 as a result of this comment.*

F.1.5 Transportation Comments and Responses

Comment:
Comment Number: 2
Section Number: 3.3 Table S-4 on the Transportation of Fuel and Waste (Page 3-3)
Comment/Basis: This section discusses the environmental data provided in Table S-4 and the applicability criteria for its use.
Recommendation: When discussing burnup, specify if bundle average or pin average burnup was considered in the criteria. (**2-4** [Pimentel, Frances A.])

Response: *The burnup levels discussed in this NUREG, such as in Appendix A, are assembly averaged burnup as presented in Hall et al. (2021-TN8084) and Kucinsk et al. (2022-TN8091) and form the basis for the radionuclide inventory provided in Appendix A. Based on this comment, this clarification was made throughout NUREG-2266 as appropriate.*

Comment:
Comment Number: 3
Section Number: 3.6.3 Number of Annual Unirradiated and Spent Accident Tolerant Fuel Shipments (Page 3-18)
Comment/Basis: Line 7 discusses how many fuel assemblies are removed from a core during an outage. For longer cycles, this will be more than one-third, therefore, recommend saying "During a typical refueling outage, between one-third to one-half of the fuel assemblies..."
Recommendation: Recommend changing lines 7 to 8 to read, "During a typical refueling outage, between one-third to one-half of the fuel assemblies..." (**2-5** [Pimentel, Frances A.])

Response: *The NRC staff acknowledges the comment. In addressing the comment, the NRC staff evaluated the environmental impacts of core reloads of one-half of the fuel assemblies in the core. The staff expanded the discussion of the transportation impacts for one-half-core reloads by adding in this NUREG updated information in Section 3, Section D.3, and Appendix E tables to account for both one-third- and one-half-core reload results. Changes were made to the NUREG based on this comment.*

Comment:
Comment Number: 6
Section Number: Appendix B, Section B.2 Cases (Page B-2)
Comment/Basis: This section references 18 events (cases) examined in NUREG/CR-6672. It would be helpful to have the description of each of these cases provided in this document for ease of reference.

Recommendation: Add a table with the description of the 18 events (cases) examined in NUREG/CR-6672. (**2-8** [Pimentel, Frances A.])

Response: *The NRC staff acknowledges the comment. The cases do not have specific names or unique descriptions and are defined by temperature and velocities as shown in Table B-2, except for the fire scenario as presented in Section 7.2.6 of NUREG/CR-6672. As indicated in the Table 7.10 Truck Accident Cases of NUREG/CR-6672, it is apparent that the descriptions and characteristics of each case are identified by velocity and temperature ranges. Changes were made to Appendix Section B.2 in this NUREG-2266 as a result of this comment to clarify the information source for the 18 events.*

Comment:
Comment Number: 7
Section Number: Appendix B, Section B.6 Particulate Release (Page B-7)
Comment/Basis: This section discusses the changes to particulate release fractions due to impact and temperature and due to fire only. The discussion does not explain why the NUREG/CR-6672 methodology assumes that for the fire-only scenario the cladding rupture could be large, and that fines in up to 1 foot (ft) of the rod could escape without filtering and that for the impact and temperature scenario, the cladding rupture opening is expected to be smaller and fines in up to 0.25 inches (in.) of the rod could escape without filtering.
Recommendation: Include a discussion as to why the cladding rupture is expected to be larger in the fire only scenario than the expected rupture opening for the impact and temperature scenario. (**2-9** [Pimentel, Frances A.])

Comment:
Page Number: Appendix B
Section Number: Multiple
Comment: The "fire only" scenario discussed in Appendix B results in a larger rod failure and particulate release than that of the "collision plus fire" scenarios. This result is counter-intuitive, but it is clarified in NUREG/CR-6672. It would be helpful to the reader if a similar clarification is provided in this report. (**3-4** [Harper, Zachary S.])

Comment:
Page Number: B-7, B-8, B-12
Section Number: B.6, B.7, B.10
Comment: Throughout Appendix B, The Report points to analyses performed in NUREG/CR-6672 and cites various cases. For example, the "fire only" scenario is identified as case 18 (per Table B-7). Per NUREG/CR-6672 Table 7.10, the "fire only" scenario is case 18 for truck accidents. Per NUREG/CR-6672 Table 7.11, the "fire only" scenario is case 20 for train accidents. Additional clarity on which accident scenario is being considered would be helpful to the reader, such as referencing the original NUREG/CR-6672 tables. (**3-11** [Harper, Zachary S.])

Response: *The comments requested clarifications and a discussion about the analyses described in NUREG/CR-6672 Section 7.2.6 concerning the truck and rail accident scenarios. The NRC staff based its evaluation presented in Appendix B of this NUREG on the analyses, methodology, and assumptions presented in NUREG/CR-6672. Rods subjected to temperature and impact are expected to fracture due to the initial impact forces, and these failures have been observed to result in small openings in the cladding of around 0.25 in. Rods subjected to* *temperature only are expected to fail by ballooning and burst as the temperature increases, resulting in larger openings in the cladding that have been observed to extend up to 1 ft axially. These estimates for rod fracture sizes are the same as used in NUREG/CR-6672. Changes were made to Section 3.5.4, Appendix B Section B.6 text and tables in this NUREG, to specifically state why only the truck transport was considered for the sensitivity analysis and Appendix Section B.6 includes the fire only accident scenario from NUREG/CR-6672.*

F.1.6 Uranium Fuel Cycle Comments and Responses

Comment: Industry would prefer that NUREG-2266 be revised to accommodate enrichments up to 10 wt% U-235; however, if this change will significantly extend its issuance, then use of 8 wt% U-235 as the bounding value is acceptable. (**2-2** [Pimentel, Frances A.])

Comment:
Comment Number: 1
Section Number: 1.3 Scope of this Study (Page 1-3)
Comment/Basis: Section 1.2 discusses how NRC staff anticipates that applicants may seek to use fuels with enrichments up to approximately 10 weight percent (wt%) uranium-235 (U-235) and higher burnup levels up to approximately 75 to 80 gigawatt days per metric ton of uranium (GWd/MTU). Then, in Section 1.3, Lines 24 thru 27 state that, "the NRC staff also assesses these impacts in this study, with increased enrichment and higher burnup levels up to 8 wt% U-235 and up to 80 GWd/MTU, respectively, on the uranium fuel cycle, transportation of fuel and waste, and decommissioning for LWRs (i.e., a bounding analysis)." There are several other instances in the document where increased enrichment is discussed as 8wt%.
Recommendation: For consistency, the definition the industry uses for LEU is between 5-10 wt% U-235. This NUREG discusses enrichments up to 8 wt% and 10 wt% and burnups up to 75 GWd/MTU and 80 GWd/MTU. If possible, industry would prefer that increased enrichment and higher burnup levels be indicated up to 10 wt% U-235 and up to 80 GWd/MTU, respectively, throughout the document as the bounds for the impacts that were assessed in this study. However, if revising this document to include enrichments up to 10 wt% U-235 will significantly the extend the issuance of this document, then 8 wt% U-235 as a bounding value is acceptable. (**2-3** [Pimentel, Frances A.])

Comment:
Page Number: 1-4, B-10
Section Number: 1.4, B.12
Comment: While it is not the stated scope of The Report, NRC analyses appear to support fuel enrichments up to 10 wt% U-235 and 85 GWd/MTU. It is specifically noted on Page 1-4 that ATF fuels with Cr-coated cladding and doped pellets demonstrated "negligible effects of ATF vs. non-ATF enrichments of 5 and 10 wt% U-235 and burnup of 62 and 80 GWd/MTU". Also, Appendix B seems to extend analyses to burnups of 85 GWd/MTU with some impact, but no changes to conclusions for the analyses considered. While these analyses may not fully resolve all issues of greater enrichment and burnup, as the work has partially started, it is recommended that the conclusions be formally extended to cover those conditions as well. (**3-5** [Harper, Zachary S.])

Response: *These comments recommended that the NRC staff's environmental evaluation of ATF with increased enrichment and higher burnup levels be extended to 10 weight percent (wt%) U-235 and to 85 GWd/MTU, respectively, based on several notes and comments made in the environmental evaluation. While industry has publicly mentioned the desire to be able to* *enrich up to 10 wt% U-235, no NRC licensee or prospective applicant or vendor has indicated plans to submit related industry analyses to the NRC to date supporting plant operations with 2-year refueling cycles for enrichment levels of up to 10 wt% U-235. The staff modified the uranium fuel cycle section (Section 2) of the NUREG-2266 for up to 10 wt% U-235 and 80 GWd/MTU.*

The principal issue during decommissioning, related to the adoption and use of ATF with increased enrichment and higher burnup, is the safe storage of the spent ATF, which must be in NRC-approved spent fuel pools and certified dry cask storage systems. Such safe storage requirements must be met before and at the time of decommissioning. These requirements take into consideration and include the criticality limits for spent fuel pool storage and thermal loading limits for NRC certified dry cask storage systems. Therefore, the staff modified the decommissioning section (Section 4) of the NUREG-2266 for up to 10 wt% U-235 and 80 GWd/MTU.

For the evaluation of truck shipment transportation accident risks of spent ATF, with increased enrichment and higher burnup, one needs the radionuclide inventory associated with up to 10 wt% U-235 enrichment and 80 GWd/MTU burnup. Radionuclide spectrum in a spent fuel assembly beyond 8 wt% U-235 is yet to be evaluated by the NRC staff or industry, and there is uncertainty regarding light-water reactor (LWR) fuel at 10 wt% U-235 enrichment operating with a 2-year refueling cycle. Regarding the analysis in Appendix B of NUREG-2266 for up to 85 GWd/MTU burnup, this is a sensitivity analysis for bounding the radionuclide release fractions from transportation accidents. The U-235 enrichment does not significantly affect the fuel thermal or mechanical performance or the cladding under transportation accident conditions. As long as the assembly averaged burnup limit of 80 GWd/MTU is retained, these calculations of fuel mechanical performance will be valid for 8% or 10% enriched fuel (see PNNL 2020 for more information). Thus, because of the limits of the available analysis of assembly averaged radionuclide inventory, the NRC staff can only justify Table S-4 as bounding for up to 8 wt% U-235 and assembly averaged 80 GWd/MTU at this time. Therefore, a site-specific transportation evaluation with a related radionuclide inventory per fuel assembly would be required in accordance with 10 CFR 51.52(b) for any license amendment request for ATF deployment and use for any enrichment levels greater than 8 wt% U-235. Changes were made in Sections 2 and 4 of NUREG-2266 as discussed above for enrichments of up to 10 wt% U-235 based on these comments.

F.2 References

10 CFR Part 51. *Code of Federal Regulations*, Title 10, *Energy*, Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.” TN250.

10 CFR Part 71. *Code of Federal Regulations*, Title 10, *Energy*, Part 71, “Packaging and Transportation of Radioactive Material.” TN301.

88 FR 60507. September 1, 2023. “Draft NUREG: Environmental Evaluation of Accident Tolerant Fuels With Increased Enrichment and Higher Burnup Levels.” *Federal Register*, Nuclear Regulatory Commission. TN9593.

Hall, R., R. Cumberland, R. Sweet, and W.A. Wieselquist. 2021. *Isotopic and Fuel Lattice Parameter Trends in Extended Enrichment and Higher Burnup LWR Fuel, Vol. I: PWR Fuel*. ORNL/TM-2020/1833, Oak Ridge, Tennessee. ADAMS Accession No. ML21088A336. TN8084.

Honnold, P., R. Montgomery, M. Billone, B. Hanson, and S. Saltzstein. 2021. *High Level Gap Analysis for Accident Tolerant and Advanced Fuels for Storage and Transportation, Spent Fuel and Waste Disposition*. SAND2021-4732, Sandia National Laboratories, Albuquerque, New Mexico. TN10172.

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NRC (Nuclear Regulatory Commission). 2020. *Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept, Interim Staff Guidance*. ATF-ISG-2020-01, Washington, D.C. ADAMS Accession No. ML19343A121. TN9603.

NRC (Nuclear Regulatory Commission). 2023. *Environmental Evaluation of Accident Tolerant Fuels with Increased Enrichment and Higher Burnup Levels, Draft Report for Comment*. NUREG-2266 Washington, D.C. ADAMS Accession No. ML23240A756. TN9589.

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NRC FORM 335 (12-2010) NRCMD 3.7	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
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Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

To minimize additional complexity for each ATF LAR environmental review, the NRC staff generically evaluated the reasonably foreseeable impacts of near-term ATF technologies with increased enrichment and higher burnup levels greater than 5 weight-percent U-235 and 62 GWd/MTU, respectively, on the uranium fuel cycle, transportation of fuel and waste, and decommissioning for LWRs (i.e., a bounding analysis). The NRC staff applied available near-term ATF performance analyses, data, and studies; information from prior NRC environmental analyses; and the assessment of other publicly available data sources and studies to complete an evaluation of ATF with increased enrichment and higher burnup levels. Based on the evaluations in this study, Table S-3 with the Continued Storage Generic Environmental Impact Statement, and the Decommissioning Generic Environmental Impact Statement would bound the deployment and use of near-term ATF for up to 10 wt% U-235 and up to 80 GWd/MTU assembly averaged burnup. Table S-4 would bound the deployment and use of near-term ATF for up to 8 wt% U-235 and 80 GWd/MTU assembly averaged burnup. Therefore, this study concludes there would be no significant adverse environmental impacts for the uranium fuel cycle, transportation of fuel and wastes, and decommissioning associated with deploying near-term ATF.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.) Accident Tolerant Fuel; Increased Enrichment; Higher Burnup; Uranium Fuel Cycle; Table S-3; Transportation of Fuel and Waste; Table S-4; Decommissioning; NRC-RADTRAN	13. AVAILABILITY STATEMENT unlimited
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