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Assessment of GT-MHR Spent Fuel Characteristics and Repository Performance

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Summary

This report describes the characteristics of spent fuel discharged from a commercial Gas-Turbine, Modular Helium Reactor (GT-MHR) and the expected performance of this spent fuel in a mined geologic repository. This assessment is based on the assumptions that the GT-MHR is fueled with low-enriched uranium and operates with a once-through fuel cycle. The fuel is in the form of small coated particles (less than 1 mm in diameter) that are consolidated into compacts, which are loaded into nuclear-grade graphite blocks. After discharge from the GT-MHR and a 1-yr period of storage in dry, water-cooled storage wells, the graphite fuel elements are loaded into multipurpose canisters that are used for 5 to 10 years of on-site interim storage, transport to the repository, and permanent disposal in the repository. This current assessment is based in large measure on previous assessments of disposal of spent fuel from a weapons plutonium-fueled GT-MHR, referred to as the Plutonium Consumption, Modular Helium Reactor (PC-MHR).

The previous assessments for the PC-MHR were performed in the 1993 to 1995 time frame. Since that time, significant changes have been proposed for the regulatory criteria that apply to disposal of spent nuclear fuel and high level waste. In addition, the Department of Energy (DOE) has updated its performance assessments for Yucca Mountain, with the most recent update published in May 2001 and titled "Yucca Mountain Science and Engineering Report." This current assessment for GT-MHR spent fuel reflects the new proposed regulatory criteria and new information provided in the 2001 DOE report.

Specific areas covered in this report include:

- A review of the previous assessments performed for the PC-MHR.
- A review and assessment of the new proposed regulatory criteria.
- A detailed description of GT-MHR spent fuel characteristics, including coated-particle fuel designs, graphite blocks, radionuclide inventories, radiotoxicity, decay heat, and proliferation resistance.
- Multipurpose canister design.
- Spent fuel handling and repository loading.
- Long-term performance of GT-MHR spent fuel in a geologic repository.

This assessment has been performed in the context of comparing GT-MHR spent fuel with that from commercial light water reactors.

A unique characteristic of GT-MHR spent fuel is the ceramic coating system consisting of pyrolytic carbon and silicon carbide. Based on the available data for the corrosion resistance of these materials, the coatings should remain intact over geologic time scales (hundreds of thousands to millions of years), even if the repository were to become permanently flooded with groundwater. Based on this and other characteristics of GT-MHR spent fuel, this waste form should be ideally suited for permanent disposal in a geologic repository.

Acronyms and Abbreviations

A_i	activity of nuclide i (curies)
A_{sm}	geometrical surface area per unit mass
b	barn (10^{-24} cm^2 – nuclear cross section unit)
CEDE	committed effective dose equivalent
Ci	curie
CSNF	commercial spent nuclear fuel
DOE	United States Department of Energy
EFPD	Effective Full Power Days
EIS	Environmental Impact Statement
EnPA	Energy Policy Act of 1992
EPA	Environmental Protection Agency
FMLR	fractional mass loss rate
GA	General Atomics
GT-MHR	Gas Turbine, Modular Helium Reactor
Gw	gigawatts
HTGR	high-temperature, gas-cooled reactor
IHI	Ingestion Hazard Index
k_{eff}	neutron multiplication factor
kw	kilowatts
LTR	layer thinning rate
LWR	light water reactor
M	Weibull parameter
MCL	maximum contaminant levels
MINATOM	Russian Federation Ministry for Atomic Energy
MPC	multipurpose canister
mrem	millirems
MTHM	metric tons of heavy metal
Mw	megawatts
N_{ppm}	nitrogen impurity content by weight (ppm)
NAS	National Academy of Sciences
NEPA	National Environmental Policy Act
NERAC	Nuclear Energy Research Advisory Committee
NRC	Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
ORNL	Oak Ridge National Laboratory
P	internal pressure inside particle coatings
P_F	failure probability of particle coating system
pCi	picocuries
PC-MHR	Plutonium Consumption, Modular Helium Reactor
PEC	permissible effluent concentration
PNL	Pacific Northwest Laboratory
PWR	pressurized water reactor
R_{SiC}	Radius to middle of SiC layer
RME	Reasonable Maximum Exposure
RMEI	Reasonably Maximally Exposed Individual
RPu	reactor-grade plutonium
t_{IRR}	irradiation time (days)

Acronyms and Abbreviations (Cont.)

t_{SiC}	SiC thickness
TOPS	Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems
WPu	weapons-grade plutonium
φ	thermal neutron flux (10^{13} $cm^{-2}s^{-1}$)
ρ	material density
σ	maximum circumferential hoop stress in SiC layer
σ_f	median fracture strength of SiC
σ_C	thermal cross section for $^{13}C(n,\gamma)^{14}C$ reaction (b)
σ_N	thermal cross section for $^{14}N(n,p)^{14}C$ reaction (b)

Subscripts

e	electrical
t	thermal

1. Introduction and Background

The U.S. Department of Energy (DOE) is currently evaluating the Yucca Mountain site for disposal of spent nuclear fuel from the current generation of commercial light-water reactors (LWRs), research reactors, naval reactors, reactor prototypes, and reactors that produced materials for nuclear weapons. The weapons programs also generated approximately 100 million gallons of liquid high-level waste that is currently being stored in underground tanks at DOE sites. After treatment and vitrification, this waste is also slated for disposal at Yucca Mountain. The Yucca Mountain Science and Engineering Report (Ref. 1) provides a comprehensive assessment of these waste forms and their performance over long-time periods after emplacement into the proposed Yucca Mountain repository.

These performance assessments show the repository should pose a negligibly small radiological risk for time periods well beyond the 10,000 years currently being proposed by the Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) for regulatory compliance, even if the waste packages were to degrade and fail at a rate much faster than expected. This good performance is due in large measure to the fact that the proposed repository would be located approximately 300 m above the water table, in a remote arid region on the Nevada Test Site that offers little potential for migration of radionuclides to the surrounding, lightly populated areas.

The DOE assessment is based on the geological record for the proposed site, which indicates there is very low probability the site characteristics will be significantly degraded by future climatic, volcanic, or seismic events. Although the DOE disposal strategy includes robust, corrosion-resistant canisters, drip shields, and other engineered barriers to provide defense in depth, the natural barriers associated with the site itself should provide more than adequate protection to the surrounding population and environment. In short, the DOE assessment makes a strong case that there are no technical impediments for direct disposal of spent fuel and high-level waste at Yucca Mountain.

Assuming a similar disposal strategy, the above conclusions should also be valid for geologic disposal of spent fuel from a future generation of advanced reactors. However, in addition to improvements in safety and economics, desirable features of advanced reactors would include: (1) fuel cycles that are highly resistant to proliferation and (2) final waste forms that are highly resistant to corrosion and degradation over geologic time scales. The latter feature would place less burden on the engineered and natural barriers of the repository for waste isolation. Additional benefits of a more robust, proliferation-resistant waste form include:

- A greater variety of sites could be considered for permanent disposal.
- Rather than having to make the case to regulators that worst-case scenarios are highly unlikely (even over geologic time scales) and do not merit analysis, it may be possible to show regulatory compliance even under nonmechanistic, worst-case conditions (e.g., repository flooding).
- A greater public acceptance for nuclear energy, which is a fundamental goal for development of advanced reactors.

1.1 The Gas Turbine, Modular Helium Reactor (GT-MHR)

The GT-MHR is currently being developed as part of an international program sponsored by DOE, the Russian Federation Ministry for Atomic Energy (MINATOM), and General Atomics (GA), with participation of other international organizations, including Framatome and Fuji Electric. Figure 1-1 shows a cross-sectional view of the GT-MHR reactor and power-conversion vessels. The GT-MHR is a passively safe, advanced reactor design that requires no active measures to mitigate the consequences of a hypothetical severe accident, including those involving complete loss of its high-pressure helium coolant. The possibility of a core meltdown is precluded through the use of refractory, coated-particle fuel, nuclear-grade graphite fuel elements with high heat capacity and thermal conductivity, and operation at a relatively low power density. The GT-MHR is designed to generate 600 Mw_t and produce 285 Mw_e (47.5% thermal efficiency) when operating with a core-outlet/turbine-inlet temperature of 850°C. Reference 2 provides a more detailed description of the GT-MHR.

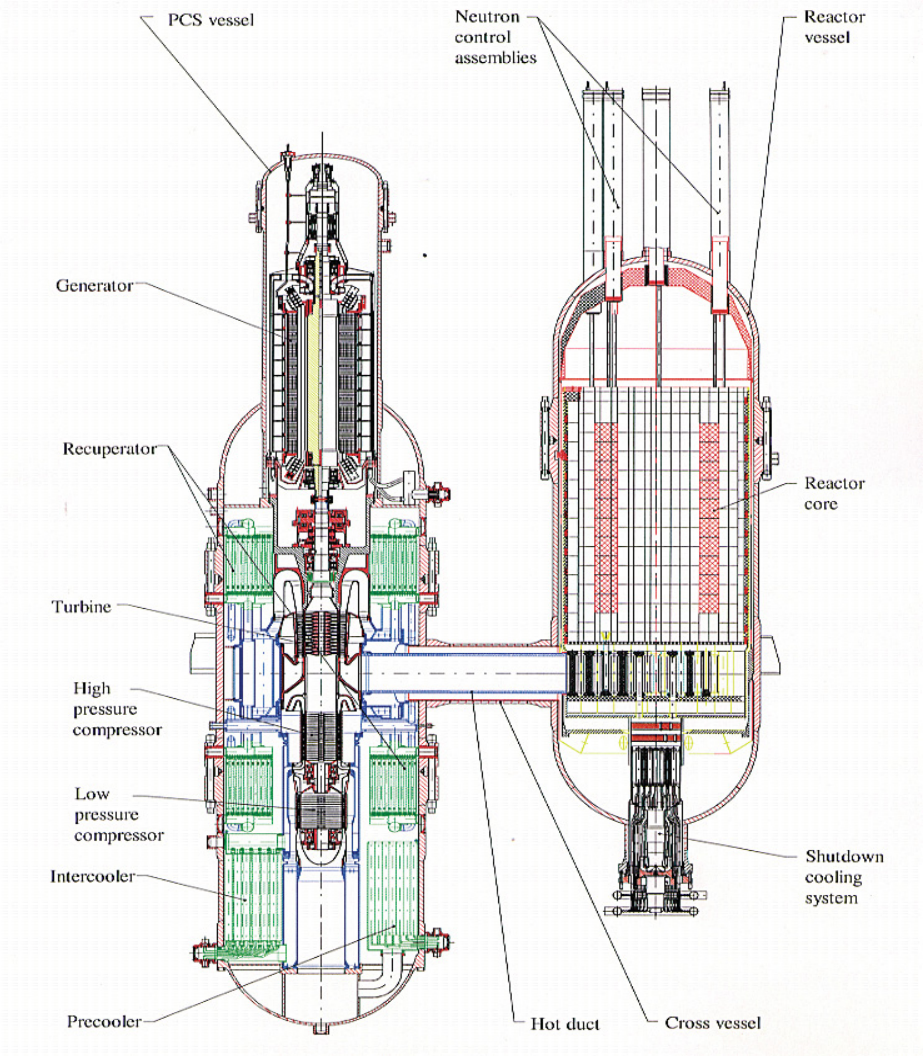

Figure 1-1. The Gas-Turbine, Modular Helium Reactor

1.2 Scope

This report describes characteristics of spent-fuel elements discharged from the GT-MHR operating on a once-through, low-enriched uranium fuel cycle and their expected long-term performance in a geologic repository. This report also provides comparisons with disposal of commercial LWR spent fuel. Specific areas addressed include applicable regulatory criteria, radionuclide inventories, radiotoxicity, decay heat loads, proliferation resistance, disposal canister design, repository loading, and long-term performance of coated fuel particles and graphite fuel elements. Much of the material presented in this report is based on previous assessments of spent-fuel disposal for a GT-MHR using weapons-grade plutonium as nuclear fuel. Results and conclusions from these previous assessments are summarized below.

1.3 Summary of Previous Work

The most comprehensive assessments of disposal of GT-MHR spent graphite fuel elements were performed in the 1993 to 1995 time frame, as part of the Plutonium Consumption, Modular Helium Reactor (PC-MHR) project. This work was performed in four phases, referred to as Phase I, Phase II, Phase II Extension, and Fiscal Year 1995 (FY-95). Results and conclusions from each phase are summarized below. More detailed descriptions of this previous work is provided in the references cited below.

Phase I

Phase 1 work is described in Ref. 3. During Phase 1, several disposal strategies were evaluated, including whole-element disposal and removal of fuel compacts from the graphite elements to reduce the volume of high-level waste. Whole-element disposal was selected as the preferred option because of advantages related to ease of implementation, proliferation risks, safeguards requirements, cost, and schedule. This recommendation was consistent with a previous study by Oak Ridge National Laboratory (ORNL) (Ref. 4), which concluded that whole elements (containing uranium and thorium fuels) appear to be satisfactory for disposal in a geological repository and should perform better than unprocessed LWR spent fuel.

Phase II

Phase II work is described in Ref. 5. In Phase II, further evaluations of whole-element disposal were performed, including an assessment of the technical criteria for use of a multi-purpose canister (MPC) to satisfy requirements for temporary dry on-site storage, transportation to the repository, and final disposal within the repository. Based on the Phase II work, it was concluded that no technical issues should preclude whole-element disposal of spent PC-MHR fuel using MPCs in a geologic repository.

Phase II Extension

Phase II Extension work is described in Ref. 6. During this phase, more comprehensive and quantitative analyses were performed that confirmed the conclusions reached during Phases I and II. Results and conclusions are summarized below:

- The high level of plutonium destruction and degradation achieved by the PC-MHR provides permanent disincentives and deterrents to the reuse of plutonium for nuclear weapons or as fuel for reactors.

- The graphite fuel elements should satisfy all regulatory requirements for permanent disposal. High-purity, nuclear-grade graphites are noncombustible by conventional standards, and oxidation of graphite and other fuel element components at repository temperatures would be negligible over geologic time periods.

- The graphite fuel elements and the ceramic coatings on the fuel particles are as-manufactured engineered barriers that provide excellent near-field containment of radionuclides and minimize reliance on the waste package and surrounding geologic media for long-term containment.

- Dilution of plutonium within the relatively large volume of PC-MHR fuel elements provides excellent resistance to diversion throughout the fuel cycle. This is accomplished without adversely impacting repository land requirements, which are determined primarily by decay heat load and not by physical volume. In fact, PC-MHR spent fuel would require about one-half of the repository land area needed for disposal of commercial LWR spent fuel, assuming the same quantity of electrical energy had been generated by the fuel discharged from each reactor type, a representative heat-load limit of 57 kw_t/acre for Yucca Mountain,[*] and 10 years between discharge of spent fuel from the reactor and repository loading.

- A conceptual MPC design was developed for PC-MHR spent fuel. The PC-MHR MPC would contain 42 fuel elements, arranged as 7 columns with 6 fuel elements per column. The overall dimensions would be identical to those for a commercial pressurized water reactor (PWR) MPC designed for 21 fuel assemblies, but the weight with spent fuel would be about one-half that of the PWR MPC The thermal, structural, shielding, criticality-control, and radionuclide-containment requirements for the PC-MHR MPC are much less demanding than those for PWR MPC, which simplifies the canister design and lowers manufacturing costs.

- For release by groundwater transport, the nuclide of most concern is C-14, because nearly all of the inventory is external to coated particles and can be released by groundwater leaching of graphite. Conservative estimates of C-14 transport and release showed the radiological consequences would be well below applicable regulatory criteria.

- The disposal cost for PC-MHR spent fuel should be a relatively small fraction of the total fuel-cycle cost and should be comparable to that for commercial LWR spent fuel, on a per unit electrical energy basis.

[*] The DOE has since revised its thermal-management strategy to include active air cooling of waste packages for a number of decades following emplacement. This strategy allows for a denser loading of waste packages into the emplacement drifts, but relies on active measures for an extended period of time to properly cool the waste packages and drift walls.

FY-95 Work

The following tasks related to PC-MHR spent fuel disposal were performed during FY-95:

- A technology-development plan was prepared to define the additional data and test programs needed to further evaluate disposal of PC-MHR spent fuel (Ref. 7).

- A computer code was developed to model near-field performance of PC-MHR waste packages in a geological repository, to estimate radionuclide transport in the geosphere, and to estimate radionuclide release to and radionuclide concentrations in the accessible environment. This computer code was named REPPER (for REPository PERformance) and is described in Ref. 8.

- A specification was prepared (Ref. 9) to define the requirements for tests to determine the residual nitrogen in unirradiated PC-MHR graphite and fuel compact matrix and the C-14 content in irradiated PC-MHR graphite and fuel compact matrix.

- Additional design work was performed on the PC-MHR MPC, and a preliminary design report was issued (Ref. 10). A reference design was selected after evaluations of five different configurations. The reference design has the same 42-element capacity of the conceptual design described in Ref. 6, and meets all regulatory requirements for storage, transportation, and disposal.

- Additional work was performed on defining repository loading strategies for PC-MHR spent fuel (Ref. 11), including evaluations of different thermal-management options and co-disposal of PC-MHR and LWR waste packages.

- Calculations were performed to estimate the radiotoxicity of PC-MHR spent fuel, and to further assess long-term stability and in-repository performance of PC-MHR spent fuel (Ref. 12). Results are summarized below:

  - PC-MHR graphite fuel elements and TRISO coatings on the plutonium fuel particles should remain stable over geologic time periods (hundreds of thousands to millions of years).

  - When normalized with respect to electrical energy production, the radiotoxicities of PC-MHR and commercial LWR spent fuel are nearly the same.

  - The REPPER code was used to calculate release of C-14 to the accessible environment. The radiological consequences of C-14 ingestion were estimated according to regulatory guidelines, and the resulting dose rates were shown to be well below regulatory criteria.

2. Regulatory Framework

Since the 1950s, several options have been evaluated in the U.S. for disposal of spent nuclear fuel and high-level waste. Based on these evaluations, disposal in mined geologic repositories has been selected as the preferred solution, because it is cost effective and should provide minimal long-term risks to the environment and public health. With its 1987 amendments to the Nuclear Waste Policy Act of 1982 (NWPA), the Congress directed DOE to investigate exclusively a proposed site at Yucca Mountain, NV for its suitability as the nation's first geologic repository. The NWPA assigned specific responsibilities to the DOE, NRC, and EPA:

- The DOE is responsible for siting, constructing, operating, and closing the repository.
- The NRC is responsible for promulgating regulations that govern the construction, operation, and closure of the repository. The NRC is also charged with licensing the repository and developing technical criteria for restrictions on waste retrievability and the use of multiple barriers for near-field containment.
- The EPA is responsible for developing public health and safety standards for releases of radioactivity from the repository.

The NWPA also assigned financial responsibility to the generators and owners for disposal of their spent nuclear fuel and high-level waste. For commercial nuclear power, the generators are assessed a fee of 0.1 cents per kw_e-hr generated by their nuclear power plants.

The EPA and NRC have been developing regulatory criteria for mined geologic repositories since the early to mid 1980s. As part of the Energy Policy Act of 1992 (EnPA), Congress directed EPA to contract with the National Academy of Sciences (NAS) for a study of the scientific basis for an EPA standard to be applied specifically at the Yucca Mountain site. After establishment of final EPA standards, the NRC has 1 year to modify its technical requirements and criteria to be consistent with the new EPA standards, and to address other issues specified in the EnPA with regard to the effectiveness of postclosure oversight of the repository.

The NAS fulfilled its obligations under EnPA with publication of its report, Technical Bases for Yucca Mountain Standards, in 1995 (Ref. 13). The NAS recommended an approach that differs in significant ways from the existing EPA standard (40CFR191[*]). For scenarios involving releases of radionuclides from the repository, the NAS recommended the standard be based on limiting the risk to exposed individuals, rather than imposing specific limits for releases of radionuclides to the accessible environment. The NAS questioned the EPA's compliance period of 10,000 years, since performance assessments showed peak risks may occur at times well beyond 10,000 years. The NAS concluded it was not possible to determine the frequency of human intrusion into the repository over long time periods, and hence recommended against risk-based calculations for assessing the adverse effects caused by human intrusion. The NAS did recommend "that the consequences of an intrusion be calculated to assess the resilience of the repository to intrusion."

[*] The designation is Title Number – Code of Federal Regulations – Part Number. Sections of the Code of Federal Regulations can be obtained from the internet site http://www.access.gpo.gov/nara/cfr/ and various other web sites maintained by the federal government.

In response to the NAS report, the EPA and NRC have published proposed regulations in the Federal Register:

- The proposed EPA regulations have been published as 64FR46976* and would be promulgated as 40CFR197, Environmental Protection Standards for Yucca Mountain, Nevada.

- The proposed NRC regulations have been published as 64FR8639 and would be promulgated as 10CFR63, Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada.

The DOE has also published proposed regulations in the Federal Register (64FR67054) regarding site suitability criteria and evaluation methods for the preclosure and postclosure periods. These regulations would be promulgated as 10CFR963, Yucca Mountain Site Suitability Guidelines, and have been structured to be consistent with the proposed EPA and NRC regulations. In its most recent performance assessment (Ref. 1), the DOE has adopted the proposed EPA and NRC regulations as a reasonable basis for judging regulatory compliance of the proposed repository at the Yucca Mountain site. For this report, it is assumed identical criteria would apply to disposal of GT-MHR spent fuel. The proposed EPA and NRC regulations are described in more detail below, with emphasis on criteria that apply to post-closure performance.

2.1. Regulatory Compliance Period

The EPA has given consideration to the NAS recommendation for a compliance period that corresponds to when peak dose occurs, even if that period extends beyond 10,000 years. However, the EPA plans to continue to use a 10,000-yr compliance period (as originally promulgated in 40CFR191) for the following reasons:

- Performance assessments for periods beyond 10,000 years are subject to large uncertainties associated with climatic conditions, the biosphere, and human activity.

- Many international geologic disposal programs use a 10,000-yr compliance period.

Although performance assessments for periods beyond 10,000 years are not required by the EPA for determining regulatory compliance, the EPA does recommend that longer-term assessments be included in the Yucca Mountain Environmental Impact Statement (EIS). By including post-10,000-yr analyses in the EIS, the EPA is intending to encourage DOE to use a robustly engineered design for its waste packages. The NRC has also adopted the 10,000-yr compliance period.

* The designation is Volume – Federal Register – Page Number. Sections of the Federal Register can be obtained from the internet site http://www.access.gpo.gov/nara/cfr/ and various other web sites maintained by the federal government.

2.2 Protection of Individuals

The EPA is proposing to use a dose-rate limit and the concept of a "reasonably maximally exposed individual"* (RMEI) for protection of individuals. The RMEI would be representative of a future population group termed "rural-residential." The RMEI is assumed to reside at a location about 20 km down gradient from the repository, near the intersection of U.S. Route 95 and Nevada State Route 373, known as Lathrop Wells. The RMEI is assumed to drink 2 liters per day of contaminated water, which is conservative and consistent with the concept of reasonable maximum exposure (RME). The RMEI is also assumed to receive exposure through consumption of contaminated food (e.g., vegetables that had been irrigated with contaminated water). In order to show compliance, the mean or median dose (whichever is higher) to the RMEI must be less than 15 mrem/yr committed effective dose equivalent (CEDE) throughout the 10,000-yr compliance period. The CEDE must consider all potential exposure pathways and is calculated using methods prescribed in Federal Guidance Reports Nos. 11 and 12 (Refs. 14 and 15, respectively). The NRC has specified a similar approach, except the CEDE is set somewhat higher at 25 mrem/yr. However, it is expected that the NRC will adopt the EPA limit when final regulations are promulgated.

2.3 Protection of the General Population

In its generic regulations (40CFR191), the EPA imposed release limits for specific radionuclides in order to discourage the selection of disposal sites near large bodies of surface water or near large sources of ground water, since these sites could rely heavily on dilution mechanisms to satisfy individual exposure requirements at the expense of increased overall exposure to the surrounding population.† However, in its proposed Yucca Mountain site-specific regulations (40CFR197), the EPA is not imposing any specific release limits, since protection of the rural-residential RMEI would likely ensure protection of the general population at an arid, lightly populated site like Yucca Mountain, especially since the aquifer under Yucca Mountain does not discharge into any large bodies of surface water. Although the proposed 40CFR197 regulations contain no specific criteria for protection of the general population, the EPA has asked DOE to consider risks to the general population as part of the normal National Environmental Policy Act (NEPA) process for evaluating technical alternatives.

2.4 Protection of Ground Water

The aquifer which flows under Yucca Mountain is currently used as a source of drinking water, and the EPA has included criteria in the proposed 40CFR197 for protection of this resource. The EPA is proposing to use the same maximum contaminant levels (MCLs) promulgated in 40CFR141, National Primary Drinking Water Regulations. The relevant MCLs are:

- 5 pCi/l for combined Ra-226 and Ra-228

* The author of this report is in no way responsible for the terminology developed and used by federal agencies.

† In other words, without specific release limits, a large number of individuals could receive a dose that was just below the limit.

- 15 pCi/l for gross alpha activity

In addition to potential contamination from the repository, the aquifer will contain radioactivity from natural sources. In order to show compliance, man-made radioactivity released from the repository, when added to naturally occurring sources, must not cause these MCLs to be exceeded. In addition, 40CFR141 imposes a 4 mrem/yr dose limit for the drinking-water pathway, assuming an individual drinks 2 liters per day of water drawn from the aquifer. This limit applies to beta and photon radiation from man-made radionuclides released from the repository.

For showing compliance with the MCLs, DOE can chose between two different methods for calculating radionuclide concentrations. These approaches are referred to as "well-capture zone" and "slice of the plume," respectively. Both approaches use the concept of a "representative volume" of ground water that would supply the needs of a future, hypothetical community. After evaluating the water needs and economic base of communities near Yucca Mountain, the EPA has selected 1285 acre-feet (1.6×10^6 m^3) as a representative volume, which is based on supplying the needs of 25 people and cultivating 255 acres of alfalfa. For the well-capture approach, it is assumed that water is pumped from a single well with an annual withdrawal equal to the representative volume, and that the well intersects the plume at the point of maximum concentration. The slice-of-the-plume approach involves analysis of a portion of the aquifer that is centered below the point of compliance, with volume equal to the representative volume. For these assessments, the EPA does not require consideration of human-intrusion scenarios (see Section 2.5) or the occurrence of low-probability natural events that can disrupt the repository (see Section 2.8).

2.5 Human-Intrusion Scenarios

The EPA is proposing that DOE evaluate a prescribed human-intrusion scenario, which involves exploratory drilling of a single borehole through the repository down to the water table to obtain water. The EPA proposes that the NRC select the time when drilling would occur. The EPA suggests that this time be within a period when a few waste packages have failed, but most of the radioactivity is still within the near-field environment. In its proposed regulations, the NRC specifies that "it shall be assumed that the human intrusion occurs 100 years after permanent closure and takes the form of a drilling event that results in a single, nearly vertical borehole that penetrates a waste package, extends to the saturated zone, and is not adequately sealed." Only releases through the borehole to the saturated zone are to be considered; hazards to the drillers or to the public from material brought to the surface by the intrusion would not be included in the assessment.

Neither the EPA nor the NRC have proposed evaluation of human-intrusion scenarios that involve intentional removal of waste packages for either beneficial or malicious purposes, even through the consequences of these scenarios could be significant, and the probability of occurrence could be relatively high because of the large quantities of fissile material (primarily plutonium) remaining in unprocessed LWR spent fuel. This scenario has sometimes been dubbed the "plutonium-mine" scenario.

2.6 Requirements for Multiple Barriers

The proposed NRC regulations state "the geologic repository shall include multiple barriers, consisting of both natural barriers and an engineered barrier system." The NRC specifies the multiple barrier approach to help address uncertainties with performance over long time periods. The NRC believes the combination of natural and engineered barriers will "enhance the resiliency of the geologic repository and increase confidence that the postclosure performance objective will be achieved."

2.7 Requirements for Performance Assessment

To demonstrate compliance with regulatory criteria, DOE will prepare a performance assessment for the proposed Yucca Mountain site for submittal to NRC as part of the licensing process. The NRC has developed specific requirements for the performance assessment:*

(a) Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site, and the surrounding region to the extent necessary, and information on the design of the engineered barrier system, used to define parameters and conceptual models used in the assessment.

(b) Account for uncertainties and variabilities in parameter values and provide the technical basis for parameter ranges, probability distributions, or bounding values used in the performance assessment.

(c) Consider alternative conceptual models of features and processes that are consistent with available data and current scientific understanding, and evaluate the effects that alternative conceptual models have on the performance of the geologic repository.

(d) Consider only events that have at least one chance in 10,000 of occurring over 10,000 years.

(e) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the performance assessment. Specific features, events, and processes of the geologic setting must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

(f) Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

* These requirements have been taken verbatim from 64FR8639.

(g) Provide the technical basis for models used in the performance assessment such as comparisons made with outputs of detailed process-level models and/or empirical observations (e.g., laboratory testing, field investigations, and natural analogs).

(h) Identify those design features of the engineered barrier system, and natural features of the geologic setting, that are considered barriers important to waste isolation.

(i) Describe the capability of barriers, identified as important to waste isolation, to isolate waste, taking into account uncertainties in characterizing and modeling the barriers.

(j) Provide the technical basis for the description of the capability of barriers, identified as important to waste isolation, to isolate waste.

2.8 Analysis of Disruptive Events

As indicated in Section 2.7, the NRC is proposing that DOE evaluate potential disruptive events that may occur during the post-closure period, provided there is a reasonable basis that the event could occur with a probability of 10^{-4} or higher over 10,000 years, i.e., the probability of occurrence is greater than 10^{-8}/yr. The DOE has identified igneous activity as the primary disruptive event that has the potential to affect long-term performance of the repository (Ref. 1). The assessment of naturally occurring disruptive scenarios differs from assessment of human-intrusion scenarios, in that the geologic record can establish a reasonable basis for estimating the probability of occurrence of natural events. For the Yucca Mountain site, DOE has estimated the mean annual probability of occurrence of an igneous disruption to be 1.6×10^{-8}, which corresponds to a probability of 1.6×10^{-4} in 10,000 years. Hence, per the proposed NRC regulations, the performance assessment would require consideration of a single igneous disruption. In Ref. 1, DOE has considered volcanic eruptions that bring waste to the surface and igneous intrusions that damage waste packages to expose radioactivity which can then migrate to the accessible environment.

For assessing compliance with the individual protection standard, the DOE weights the calculated dose to an individual according to the probability that igneous activity would have occurred during that individual's existence. For example, an individual living near the repository 10,000 years after closure is 100 times more likely to receive radiation exposure as the result of igneous activity than an individual living at the same location 100 years after closure. Use of a probability-weighted dose emphasizes the overall risk in terms of both the likelihood and consequences of the volcanic event, and is consistent with guidance from the NRC (Ref. 16).

Two other disruptive scenarios, repository flooding and nuclear criticality, have been the focus of scientific interest, primarily because these events could significantly degrade performance of the repository's natural and engineered barriers. The DOE and others have determined that nuclear criticality is highly improbable and well below the screening threshold of less than 1 chance in 10,000 in the first 10,000 years following emplacement. The DOE and others have also concluded that repository flooding is not scientifically credible, primarily because of the geologic record of the site. In Ref. 1, the DOE provides a detailed discussion of these scenarios and their reasons for not including them in their performance assessments.

3. Characteristics of GT-MHR Spent Fuel

The GT-MHR fuel element and its components are shown in Fig. 3-1. The manufacturing processes and quality-control methods are described in detail in Ref. 17. Detailed specifications and acceptance criteria, including allowable defect fractions, are given in Ref. 18. The following sections provide descriptions of the coated fuel particles, fuel compacts, and fuel-element graphite blocks for the GT-MHR conceptual design described in Ref. 19.

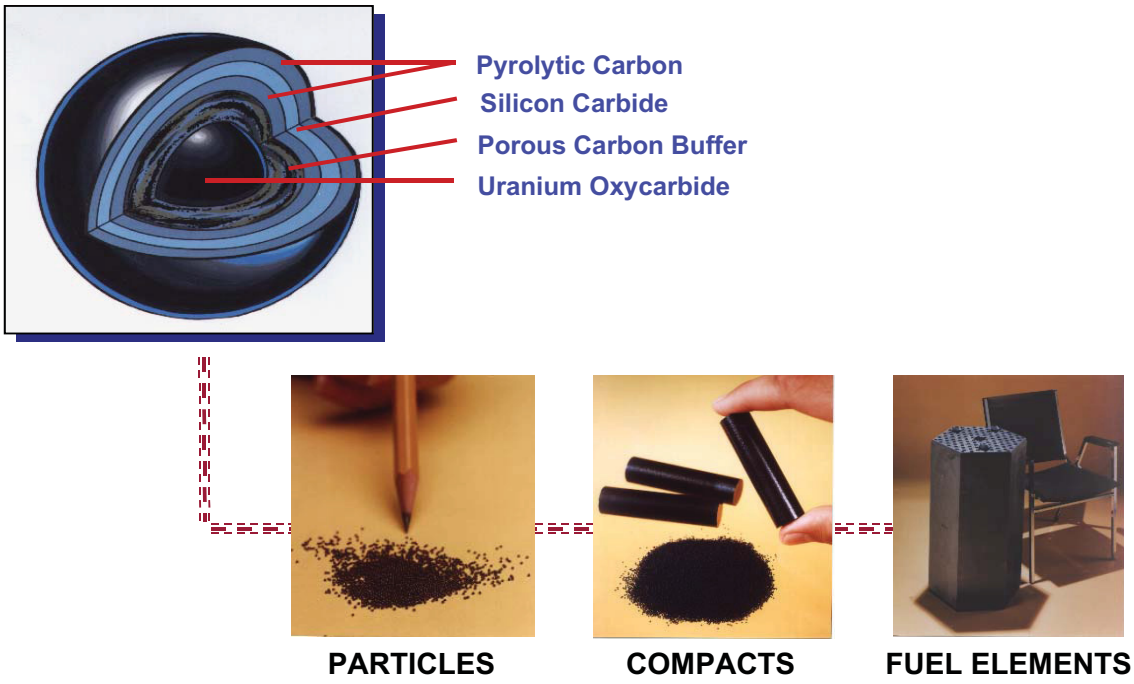

Figure 3-1. GT-MHR Fuel Element Components

3.1 Coated Fuel Particles

As shown in Fig. 3-1, the fuel for the GT-MHR consists of microspheres of uranium oxycarbide that are coated with multiple layers of pyrocarbon and silicon carbide. The GT-MHR core is designed to use a blend of two different particle types; a fissile particle that is enriched to 19.8% U-235 and fertile particle with natural uranium (0.7% U-235). The fissile/fertile loading ratio is varied with location in the core, in order to optimize reactivity control, minimize power peaking, and maximize fuel cycle length. The buffer, inner pyrolytic carbon (IPyC), silicon carbide (SiC), and outer pyrolytic carbon (OPyC) layers are referred to collectively as a TRISO* coating. The coating system can be viewed as a miniature pressure vessel that provides containment of radionuclides and gases. As discussed later in Section 6, this coating system is also an excellent engineered barrier for long-term retention of radionuclides in a repository

* TRISO is an acronym for TRI-material, ISOtropic, with the materials being low-density pyrolytic carbon (buffer), high density pyrolytic carbon (IPyC and OPyC), and SiC.

environment. Coated particle design parameters are given in Table 3-1. Fuel quality and performance specifications are given in Table 3-2.

Table 3-1. Coated Particle Design Parameters

	Fissile Particle	Fertile Particle
Composition	$UC_{0.5}O_{1.5}$	$UC_{0.5}O_{1.5}$
Uranium enrichment, %	19.8	0.7 (Natural Uranium)
Dimensions (µm)		
Kernel Diameter	350	500
Buffer thickness	100	65
IPyC thickness	35	35
SiC thickness	35	35
OPyC thickness	40	40
Particle diameter	770	850
Material Densities (g/cm^3)		
Kernel	10.5	10.5
Buffer	1.0	1.0
IPyC	1.87	1.87
SiC	3.2	3.2
OPyC	1.83	1.83
Elemental Content Per Particle (µg)		
Carbon	305.7	379.9
Oxygen	25.7	61.6
Silicon	104.5	133.2
Uranium	254.1	610.2
Total particle mass (µg)	690.0	1184.9
Design burnup (% FIMA)[a]	26	7

Note
a. FIMA is an acronym for Fissions per Initial Metal Atom.

Table 3-2. Fuel Quality and Performance Specifications

	Maximum Expected Value	Design Upper Limit
As-Manufactured Fuel Quality Defect Fractions		
Heavy metal contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or permeable IPyC	$\leq 4.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Missing or defective OPyC	$\leq 1.0 \times 10^{-4}$	$\leq 1.0 \times 10^{-3}$
Allowable Core-Average Failure Fractions		
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$
Accidents	$\leq 1.5 \times 10^{-4}$	$\leq 6.0 \times 10^{-4}$

The functions of the fuel kernel and coating layers during operation of the GT-MHR are described below:

Fuel Kernel

The oxycarbide kernel composition was selected for the GT-MHR primarily because of its ability to perform well at relatively high burnup. The carbide component of the kernel undergoes oxidation to getter excess oxygen released during fission. If the carbide component were not present, excess oxygen would react with carbon in the buffer to form carbon monoxide (Ref. 20). High levels of carbon monoxide can lead to failure of the coating system by overpressurization (Ref. 21) and kernel migration (Ref. 22). The oxide component of the kernel is highly effective at retaining many radionuclides that can chemically attack or diffuse through the coating layers (e.g., lanthanides and strontium, respectively).

Buffer

The buffer is deposited over the kernel and consists of low-density, porous pyrocarbon. The buffer attenuates fission fragments that recoil from the kernel and provides sufficient void space to accommodate gases, including gaseous fission products and CO. The buffer also acts as a sacrificial layer to accommodate potential kernel migration and swelling and isolates the kernel from load-bearing layers of the coating system.

IPyC Layer

The high-density IPyC layer serves to protect the kernel and buffer from chemical attack by chlorine compounds, which are generated as byproducts during deposition of the SiC layer. The IPyC layer also provides a smooth surface for deposition of the SiC layer and delays transport of radionuclides to the SiC layer. The IPyC layer shrinks with the accumulation of fast neutron fluence, which helps to maintain the SiC layer in compression, provided the bond between the IPyC and SiC layers remains strong and continuous during irradiation.

SiC Layer

The SiC layer is deposited under conditions to produce a high-density, high-strength coating with a fine-grain microstructure. This layer provides the primary structural support to accommodate stresses generated by internal gas pressure and irradiation-induced dimensional changes of the pyrocarbon layers. The SiC layer provides an impermeable barrier to gaseous, volatile, and most metallic fission products during normal operation and hypothetical accidents. Dimensional changes of the SiC are very small during irradiation, and it is considered to be dimensionally stable.

OPyC Layer

The high-density OPyC layer protects the SiC layer from mechanical damage that may occur during fabrication of fuel compacts and fuel elements, and provides a bonding surface for the compact matrix. The OPyC layer also shrinks during irradiation, which helps to maintain the SiC layer in compression. The OPyC layer prevents the release of gaseous fission products, if both the IPyC and SiC layers are defective or fail in service.

3.2 Fuel Compacts

Each fuel compact is a mixture of fissile, fertile, and graphite shim particles bonded together with a carbonaceous matrix into a rod-shaped compact with dimensions 12.45 mm (0.49 in.) in diameter and 49.3 mm (1.94 in.) in length. The fuel compacts are stacked in the blind fuel holes of the graphite fuel element. Graphite plugs are cemented into the tops of the fuel holes to enclose the stacked compacts. Because of sorption mechanisms, the fuel compacts can provide an additional barrier to the release of metallic fission products. Fuel compact design parameters are given in Table 3-3.

Table 3-3. Fuel Compact Design Parameters

Diameter, mm	12.45
Length, mm	49.3
Volume, cm^3	6.0
Shim particle composition	H-451 or TS-1240 graphite
Shim particle size	99 wt % < 1.19 mm
	95 wt % < 0.59 mm
Shim particle density (g/cm^3)	1.74
Binder type	Petroleum pitch
Filler	Petroleum derived graphite flour
Matrix density (g/cm^3)	0.8 to 1.2
Volume fraction occupied by matrix	0.39
Volume fraction occupied by shim particles in an average compact[a]	0.41
Volume fraction occupied by fissile particles in an average compact[a]	0.17
Volume fraction occupied by fertile particles in an average compact[a]	0.03
Number of fissile particles in an average compact[a]	4310
Number of fertile particles in an average compact[a]	520
Mass of carbon in an average compact,[a,b] g	6.62

Notes
a. Values for an average compact are determined by assuming heavy metal (uranium) is distributed uniformly in the reactor core.
b. This value excludes carbon in the layers of the coated particles. For an average compact, there is an additional 1.32 g of carbon associated with fissile particles and an additional 0.20 g of carbon associated with fertile particles.

3.3 Fuel-Element Graphite Blocks

The standard GT-MHR fuel-element graphite block and the arrangement of fuel holes, coolant holes, and lumped burnable poison* (LBP) holes is shown in Figure 3-2. The graphite blocks are fabricated from high-purity, nuclear-grade H-451 graphite. Each block is a right hexagonal prism with dimensions 794 mm (31.2 in.) in length and 360 mm (14.2 in.) across the flats of the hexagonal cross section. Fuel and coolant holes run parallel through the length of the block in a regular triangular pattern of nominally two fuel holes per coolant hole. The pitch of the coolant and fuel-hole array is 18.8 mm (0.74 in.). The minimum web thickness between a coolant hole and fuel hole is 4.5 mm (0.18 in.). This web provides an additional barrier to release of metallic fission products. Design parameters for the standard fuel element are given in Table 3-4. In addition to standard fuel elements, the GT-MHR active core contains fuel elements with a single, larger diameter channel (3.75 to 4.0 in.) to allow insertion of additional poison for reserve shutdown capability. Control rods in the active core are used only during startup and are fully withdrawn before the reactor becomes critical. Figure 3-3 shows the GT-MHR annular-core arrangement of fuel elements and reflector blocks. The active core contains 102 columns of fuel elements stacked 10 elements high, for a total of 1020 elements.

3.4 Fuel Cycle

For the equilibrium fuel cycle, one-half of the core (510 fuel elements) is reloaded every 417 full-power days, corresponding to an equilibrium residence time of 834 effective full-power days (EFPD) for each fuel element.† Each reload segment contains 1746 kg of low-enriched uranium and 507 kg of natural uranium. With a capacity factor of 85%, the GT-MHR would discharge 510 fuel elements every 16 months, or an average of about 380 elements per calendar year. Over its 60-yr plant life, a single GT-MHR module would discharge a total of about 23,000 spent-fuel elements. At discharge, an average fuel element has generated approximately 0.637 Mw_e-yr of energy.

* B_4C is used as lumped (or fixed) burnable poison to control reactivity. Compacts containing coated B_4C and graphite shim granules are inserted into holes designated for lumped burnable poison, which are located near the corners of the block.

† For the PC-MHR, the equilibrium fuel residence time was slightly shorter at 779 EFPD.

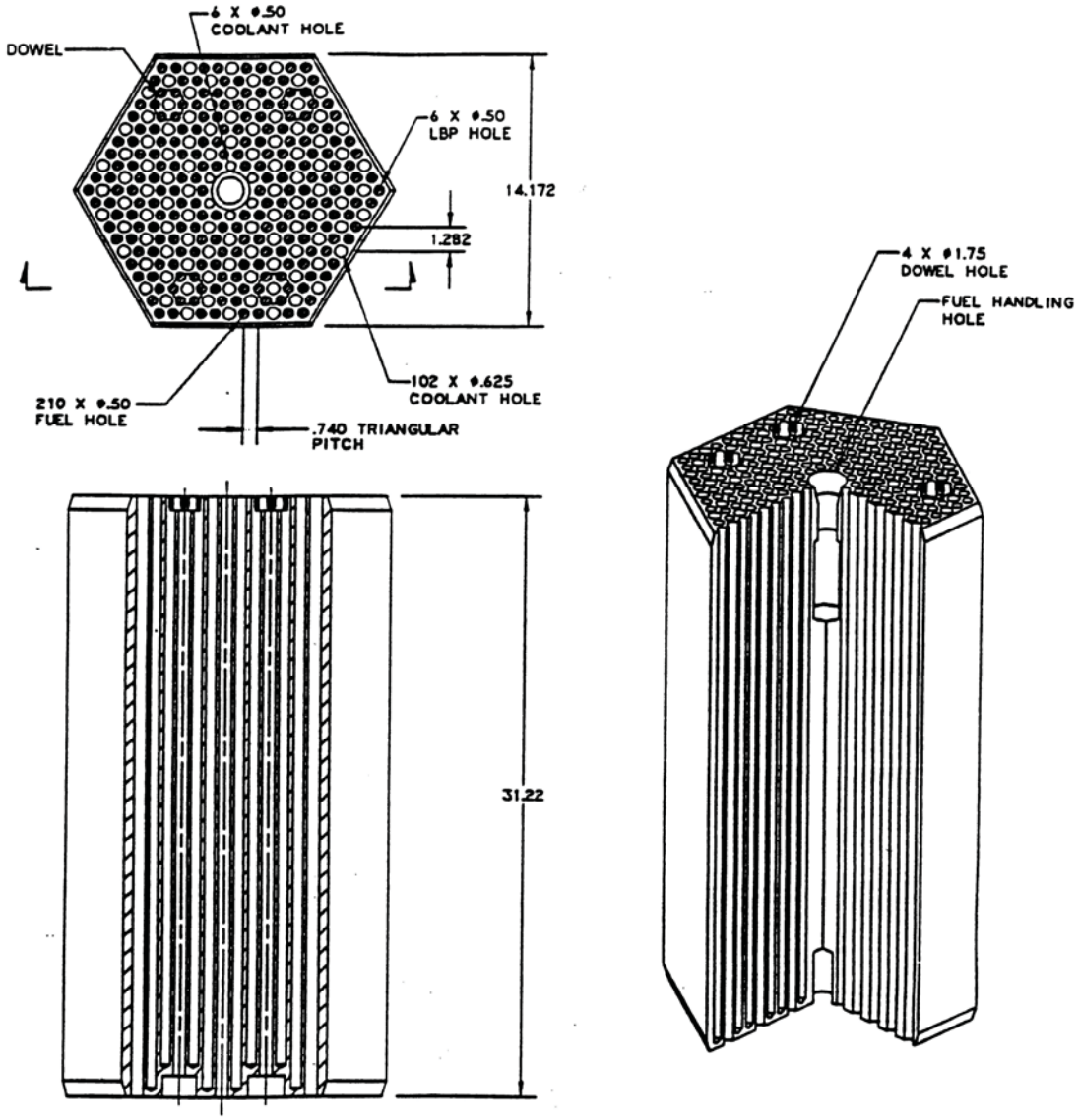

Figure 3-2. GT-MHR Standard Fuel Element (dimensions shown are in inches)

Table 3-4. GT-MHR Standard Fuel Element Design Parameters

Shape	Hexagonal Prism
Type of graphite	Nuclear Grade H-451
Mass of graphite per element	90 kg
Dimensions	794 mm (31.2 in.) in length
	360 mm (14.2 in.) across flats of hexagon
Volume[a]	0.0889 m^3
Total number of fuel holes	210
Number of fuel holes under dowels	24
Fuel hole diameter	12.7 mm (0.5 in.)
Fuel hole length	752.6 mm (29.63 in.) under dowels
	781.5 mm (30.77 in.) not under dowels
Number of fuel compacts per fuel hole	14 for holes under dowels
	15 for holes not under dowels
Number of fuel compacts per element	3126
LBP holes per element	6
LBP hole diameter	12.7 mm (0.5 in.)
LBP hole length	781.5 mm (30.77 in.)
Total number of coolant holes	108
Coolant hole diameter	15.88 mm (0.625 in.) for larger holes
	12.7 mm (0.5 in.) for the 6 smaller holes near the center of the block
Pitch of coolant/fuel-hole array	18.8 mm (0.74 in.)
Total mass of an average fuel element[b,c]	122 kg
Mass of carbon in an average fuel element[b,d]	110.7 kg
Mass of low-enriched uranium fuel in an average fresh fuel element[b]	3.43 kg
Mass of natural uranium fuel in an average fresh fuel element[b]	0.995 kg
Number of fissile particles in an average fuel element[b]	1.35×10^7
Number of fertile particles in an average fuel element[b]	1.63×10^6
Electrical energy generated by an average fuel element at discharge[b]	0.637 Mw_e-yr

Notes

a. Calculated assuming a solid hexagonal prism with all fuel and coolant holes filled, i.e., this is the physical volume a fuel element would occupy.
b. Values for an average fuel element are determined by assuming heavy metal (uranium) is distributed uniformly in the reactor core.
c. This value includes graphite and fuel compacts, but excludes lumped burnable poison.
d. This value excludes carbon in the layers of the coated particles. For an average fuel element, there is an additional 4.13 kg of carbon associated with fissile particles and an additional 0.62 kg of carbon associated with fertile particles.

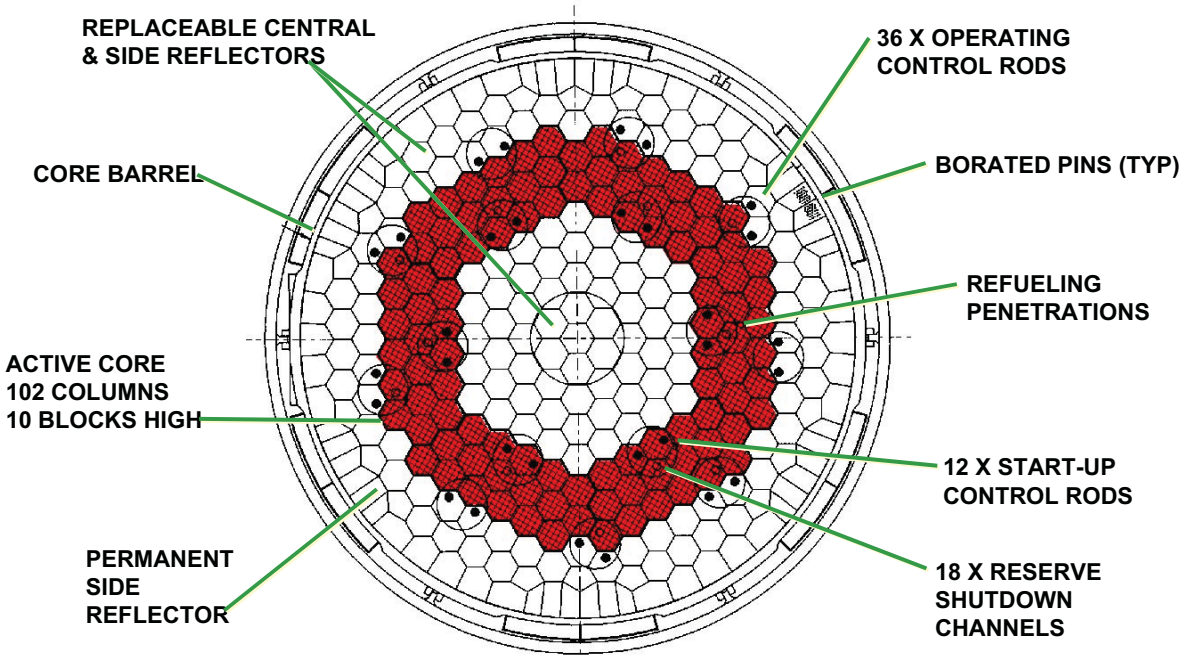

Figure 3-3. GT-MHR Annular Core Arrangement

3.5 Radionuclide Inventories

The GT-MHR spent fuel radionuclide inventory can be categorized into activation products, fission products, and actinides. The activation products are generated primarily as the result of nuclear reactions with impurities in graphite and fuel compacts. The actinide inventories were obtained from the three-dimensional burnup calculations described in Ref. 23. The fission product inventories were calculated using the GARGOYLE code (Ref. 24). The ORIGEN code (Ref. 25) was used to calculate the activation product inventory, since GARGOYLE lacks the necessary activation cross sections for all impurity elements. The activation product inventories were assumed to be the same as those calculated for the PC-MHR (Ref. 12), since the impurity concentrations, neutron fluxes, and fuel-residence times are nearly the same for both reactor cores. The impurity levels assumed for activation analysis are given in Table 3-5. With the exception of nitrogen, these impurity levels are the same as those used by the British to calculate activation product inventories in graphite from decommissioned Magnox and Advanced Gas Reactors (Ref. 26). For the present analysis, the nitrogen content was assumed to be at the design limit of 100 ppm, in order to obtain a conservative estimate of C-14 inventory.

As described in Ref. 1, the DOE has performed a comprehensive screening of radionuclides in commercial spent nuclear fuel and the other waste forms slated for disposal in the Yucca Mountain repository. The radionuclides were screened to include those that account for at least 95 percent of the potential dose to the RMEI, with consideration given to decay

chains, transport characteristics, release scenarios, and release pathways. Figure 3-4 shows all of the radionuclides considered in the repository performance assessment, segregated according to those that are fission products and those that are present in actinide decay chains.* Table 3-6 describes the nuclides shown on Fig. 3-4 with regard to their expected transport behaviors and potential contributions to dose and groundwater contamination for the nominal, volcanic, and human-intrusion release scenarios.

Table 3-5. Impurity Levels Assumed for Activation Analysis of GT-MHR Fuel Elements

Element	Concentration, ppm	Grams Per Fuel Element
Lithium	0.04	4.4×10^{-3}
Beryllium	0.02	2.2×10^{-3}
Nitrogen	100	11.1
Chlorine	3	0.33
Calcium	30	3.3
Manganese	0.15	1.7×10^{-2}
Iron	9	1.0
Cobalt	0.36	4.0×10^{-2}
Nickel	3.5	0.39
Zinc	0.6	6.6×10^{-2}
Molybdenum	1.3	0.14
Silver	0.001	1.1×10^{-4}
Cadmium	0.06	6.6×10^{-3}
Tin	0.5	5.5×10^{-2}
Barium	1.0	0.11
Europium	0.005	5.5×10^{-4}

* Although C-14 and U-232 are technically not fission products, they are included in this group because they are not members of actinide decay chains.

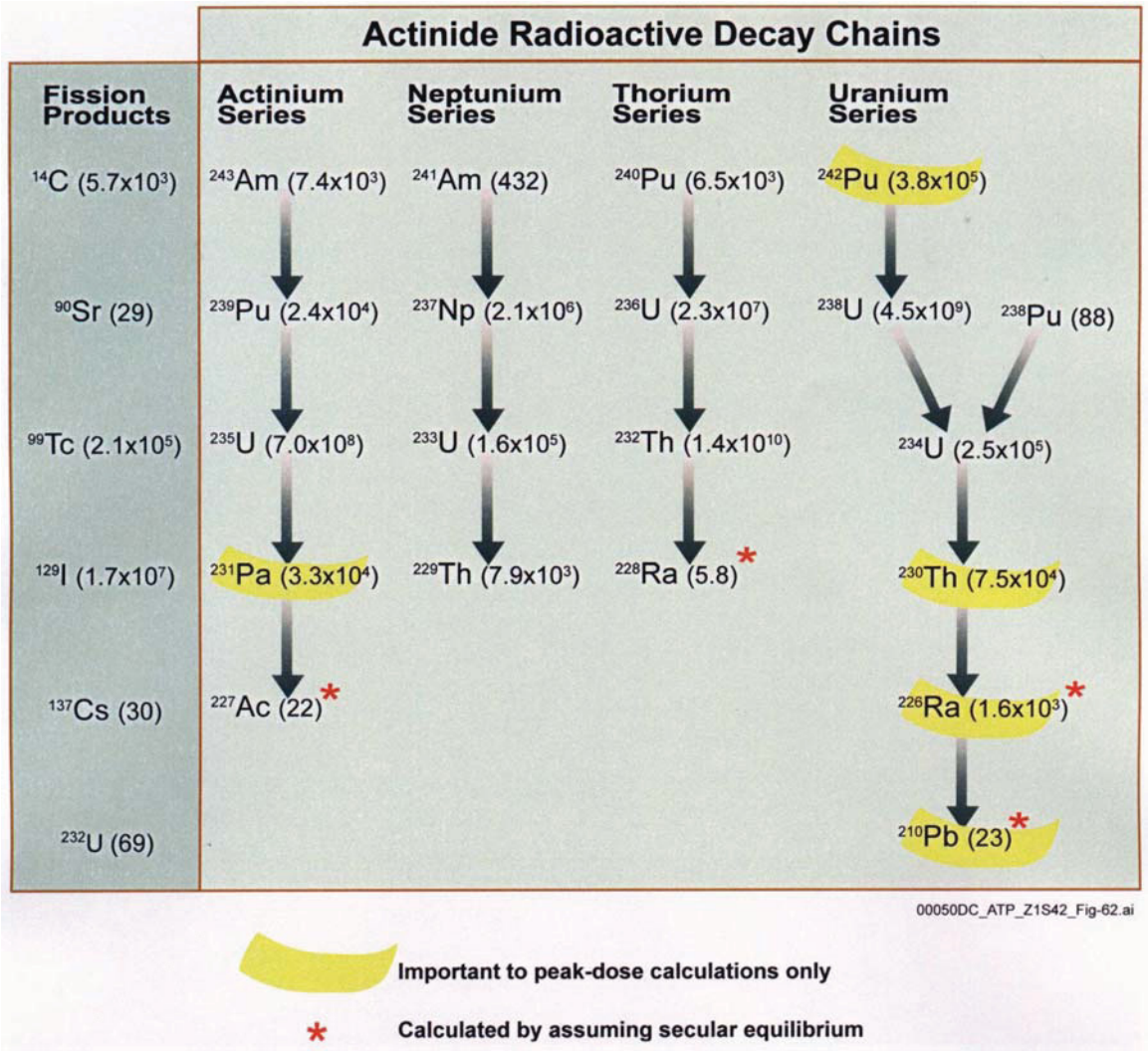

Figure 3-4. Radionuclides Considered in the Performance Assessment of the Yucca Mountain Repository (nuclide half-lives in yr are given in parentheses; figure reproduced from Ref. 1)

Table 3-6. Radionuclides Considered for the Different Release Scenarios (Ref. 1)

Nuclide	Half-Life (y)	Release Scenario			Comments
		Nominal	Human Intrusion	Disruptive Igneous Event	
Slightly to Nonsorbing Nuclides That Transport Easily in Groundwater					
C-14	5730	X	X		
I-129	1.57×10^7	X	X		
Tc-99	2.13×10^5	X	X		
Nuclides That Are Moderately Sorbed During Groundwater Transport					
Np-237	2.14×10^6	X	X		
U-232	69	X	X	X	
U-233	1.59×10^5	X	X	X	
U-234	2.46×10^5	X	X	X	
U-235	7.04×10^8				Not important to dose, but included to track Ac-227.
U-236	2.34×10^7	X	X		
U-238	4.47×10^9	X	X		
Nuclides That Are Strongly Sorbed During Groundwater Transport					
Ac-227	21.8	X	X	X	
Am-241	432.7	X	X	X	
Am-243	7370	X	X	X	
Cs-137	30.2		X	X	
Pa-231	3.28×10^4	X	X	X	Not important to dose, but included to track Ac-227.
Pb-210	22.3	X	X	X	
Pu-238	87.7	X	X	X	
Pu-239	2.41×10^4	X	X	X	
Pu-240	6560	X	X	X	
Pu-242	3.75×10^5	X	X	X	
Ra-226	1600	X	X	X	Considered because of groundwater protection requirements.
Ra-228	5.76				
Sr-90	29.1		X	X	
Th-229	7300	X	X	X	
Th-230	7.54×10^4	X	X	X	
Th-232	1.40×10^{10}				Considered because it generates Ra-228.

Inventories at discharge for several of these nuclides are given in Table 3-7. For comparison, inventories for spent fuel discharged from a typical PWR are also given. The PWR inventories were obtained from Ref. 27 and are for fuel irradiated to a burnup of 33,000 Mw_t-d per metric ton of heavy metal (MTHM) and residence time of 876 EFPD.

Table 3-7. Discharge Inventories of Radionuclides Important to Assessments of Repository Performance

Nuclide	Half-Life (y)	Specific Activity (Ci/g)	GT-MHR Ci per Fuel Element	Ci per Mw_e-yr	
				GT-MHR	PWR
Am-241[a]	432.7	3.44	1597	2508	3518
Am-243	7370	0.20	0.145	0.23	0.60
C-14	5730	4.46	0.128 (100 ppm N)	0.20	8.34×10^{-3}
			0.043 (30 ppm N)	0.068	
Cs-137	30.2	86.5	1257	1973	3675
I-129	1.57×10^{7}	1.73×10^{-4}	3.32×10^{-4}	5.21×10^{-4}	1.26×10^{-3}
Np-237	2.14×10^{6}	7.05×10^{-4}	4.13×10^{-3}	6.48×10^{-3}	0.018
Pu-238	87.7	17.1	29.5	46.3	126
Pu-239	2.41×10^{4}	6.21×10^{-2}	2.15	3.38	11.0
Pu-240	6560	0.227	4.49	7.05	16.3
Pu-242	3.75×10^{5}	3.93×10^{-3}	0.035	0.055	0.047
Sr-90	29.1	137	1042	1636	2638
Tc-99	2.13×10^{5}	1.70×10^{-2}	0.17	0.27	0.49
U-235	7.04×10^{8}	2.16×10^{-6}	4.64×10^{-4}	7.28×10^{-4}	5.76×10^{-4}
U-236	2.34×10^{7}	6.47×10^{-5}	5.22×10^{-3}	8.20×10^{-3}	9.03×10^{-3}
U-238	4.47×10^{9}	3.36×10^{-7}	1.18×10^{-3}	1.86×10^{-3}	0.011

Note
a. The discharge inventory for Am-241 includes the inventory of its parent nuclide, 14.4-yr Pu-241, since nearly all of the Pu-241 will decay to Am-241 within 100 years after emplacement of spent fuel into the repository.

With the exception of C-14, Pu-242, and U-235, the GT-MHR inventories in Table 3-7 are significantly lower than the PWR inventories, when normalized with respect to the electrical energy generated by the spent fuel. The differences in discharge inventories arise from a number of factors:

- The thermal efficiency of the GT-MHR is significantly higher than that of the PWR (47.5% vs. 32%).

- There are significant differences in the fuel cycle. Because the PWR uses much lower enriched fuel, it requires 33.6 MTHM/Gw_e-yr, whereas the GT-MHR requires only 6.95 MTHM/Gw_e-yr. The higher enriched fuel for the GT-MHR results in higher discharged quantities of U-235.

- Because of its higher-burnup fuel cycle and high thermal efficiency, the GT-MHR produces much less plutonium than the PWR. Using the values in Table 3-7 and neglecting 14.4-yr Pu-241, the PWR produces about 2.5 times the plutonium produced by the GT-MHR (268 kg Pu/Gw_e-yr vs. 102 kg Pu/Gw_e-yr). The Pu in PWR spent fuel is also of higher quality than that in GT-MHR spent fuel (66% Pu-239 vs. 53% Pu-239). The high burnup of GT-MHR spent fuel shifts the plutonium isotopics to higher mass numbers, which is why the Pu-242 inventory in GT-MHR spent fuel is

slightly higher than that in PWR spent fuel. The proliferation resistance of GT-MHR spent fuel is discussed in more detail in Section 3.8.

- There are differences in the effective fission yields. Compared with the PWR, the GT-MHR neutron energy spectrum is less thermalized and, because of its lower plutonium inventory, the percentage of fissions occurring in plutonium isotopes is significantly lower for the GT-MHR.

The higher C-14 inventory for the GT-MHR merits further discussion, since C-14 is a relatively long-lived beta emitter that can pose a potential hazard if ingested in significant quantities. Also, the bulk of the inventory is within the graphite blocks and compact matrix, making it much more susceptible to release than the inventory contained within the ceramic coatings of the fuel particles. Carbon-14 is generated from nuclear reactions with nitrogen, which is present as an impurity in graphite and compact matrix material, and from activation of C-13, which has a natural abundance of 1.1%. The nuclear reactions are $^{14}N(n,p)^{14}C$ and $^{13}C(n,\gamma)^{14}C$, respectively. The inventory in a single fuel element can be estimated from the following expression:

$$A_{C-14} = 5.97 \left(7.14 \times 10^{-8} N_{ppm} \sigma_N + 8.46 \times 10^{-4} \sigma_C\right) \varphi \, t_{IRR} \quad , \qquad (3\text{-}1)$$

where A_{C-14} is the C-14 inventory in curies, N_{ppm} is the nitrogen impurity content by weight in parts per million (ppm), σ_N is the thermal cross section in barns (b) for the $^{14}N(n,p)^{14}C$ reaction, σ_C is the thermal cross section in barns for the $^{13}C(n,\gamma)^{14}C$ reaction, φ is the thermal neutron flux in 10^{13} $cm^{-2}s^{-1}$, and t_{IRR} is the irradiation time in days. The cross sections σ_N and σ_C are 0.683 b and 3.3×10^{-4} b, respectively for the GT-MHR neutron energy spectrum (Ref. 28). The time-averaged thermal neutron flux experienced by an average-power fuel element in the GT-MHR is approximately 5×10^{13} $cm^{-2}s^{-1}$. For the assumed nitrogen content of 100 ppm, the $^{14}N(n,p)^{14}C$ reaction accounts for about 95% of the C-14 inventory. For the Fort St. Vrain HTGR, the nitrogen impurity content in graphite fuel elements was estimated to be about 30 ppm (Ref. 28). Using Eq. (3-1), the calculated C-14 inventory per fuel element is 0.043 Ci with 30 ppm of nitrogen and 0.128 Ci with 100 ppm of nitrogen. The former value is probably more representative for comparison with the PWR C-14 inventory. Nonetheless, the GT-MHR C-14 inventory would still be about a factor of 8 higher than that for the PWR on a per unit electrical energy basis. The higher C-14 inventory for the GT-MHR is a simply a direct result of the large quantity of graphite in the GT-MHR core.* However, C-14 has very little impact on repository performance, even for time periods well beyond the 10,000-yr compliance period (see Section 6).

The potential risk posed by this C-14 inventory can also be judged to some extent by comparing it with the NRC criteria specified in 10CFR61 for low-level radioactive wastes that can be disposed in near-surface facilities (e.g., the Barnwell, SC disposal site). The criteria given in Table 3-7 for Class C low-level waste were taken from 10CFR61 and are appropriate for comparison with the radionuclide inventory in GT-MHR graphite. For the GT-MHR, the bounding estimate for C-14 inventory is 0.128 Ci per fuel element (see Table 3.7), which

* In addition to its roles as a neutron moderator and reflector, the graphite blocks in the GT-MHR core have an important safety function, with their large volume keeping the power density relatively low and their large heat capacity preventing rapid temperature increases during loss-of-circulation or loss-of-coolant accidents.

corresponds to a concentration of 1.44 Ci/m^3 that is more than a factor of five below the limit specified in 10CFR61. The 10CFR61 criteria are satisfied with much greater margins for the other nuclides listed in Table 3-8, even if fuel failure and radionuclide release to the graphite was significantly higher than expected. Hence, the graphite blocks could be disposed as Class C waste, if a disposal strategy were adopted that includes separation of the fuel compacts from the graphite. This strategy would reduce the physical volume of high-level waste by about a factor of five. However, as was concluded in previous assessments for the PC-MHR, a better approach from a technical and engineering perspective is to retain the graphite as part of the disposal package. At present, there are no compelling technical reasons to consider alternatives to whole-element disposal, especially since decay-heat load is still a dominant factor for developing repository loading strategies. If reducing the high-level waste volume were to become a critical issue for continued development of the GT-MHR, then a disposal strategy could be adopted that involves compact separation, and possibly even compact deconsolidation to further reduce the volume.

Table 3-8. Radionuclide Concentration Limits for Class C Low-Level Waste

Radionuclide	10CFR61 Concentration Limit
C-14	8 Ci/m^3
Tc-99	3 Ci/m^3
I-129	0.08 Ci/m^3
Sr-90	7000 Ci/m^3
Cs-137	4600 Ci/m^3
Ni-63	700 Ci/m^3
Pu-241	3500 nCi/g
Cm-242	20,000 nCi/g
α-emitting transuranics with half-life greater than 5 yr	100 nCi/g

3.6 Radiotoxicity Assessment

For disposal of high-level nuclear waste in a geologic repository, groundwater transport is the most likely pathway for radionuclide release to the accessible environment, and ingestion of contaminated groundwater (either directly from drinking or indirectly from eating) is the most likely pathway for exposure of individuals. One measure of the radiological hazard of a given waste form is the Ingestion Hazard Index (IHI), which is defined according to

$$IHI = \sum_{i}^{N} A_i / PEC_i \quad , \tag{3-1}$$

where A_i is the activity of nuclide i in Ci and PEC_i is the Permissible Effluent Concentration of nuclide i in Ci per m^3 of water, which gives m^3 of water as the units for IHI. The values for PEC are obtained from 10CFR20. Assuming all of the radioactivity in spent fuel could be dissolved in water and that this water was to be released to the environment, the IHI provides an estimate of the dilution volume required to satisfy the criteria for release. Typically, extremely large dilution volumes on the order of 10^{15} to 10^{16} gallons of water would be required to dilute the radioactivity in a quantity of spent fuel that has generated one Gw_e-yr of electricity. The IHI provides a

relative measure of the radiotoxicity of a given quantity of waste, but it is not an appropriate parameter for judging long-term performance of a waste form in a geologic repository, since it does not account for near-field containment and transport through the geosphere. For example, a highly leachable, low-radiotoxicity waste would likely perform much worse than a high-radiotoxicity waste that is contained within multiple barriers that are highly resistant to corrosion. Also, unless the spent fuel is reprocessed, the IHI value is largely determined by isotopes of plutonium and other transuranic actinides which, for the most part, are highly sorbing on geologic media (see Table 3-6).

Nonetheless, the IHI is sometimes a useful parameter for comparing spent fuel from different reactor types. For these comparisons, it is appropriate to normalize the IHI with respect to the benefit derived from generating the spent fuel, which would be electrical energy for reactors that produce electricity.* Figure 3-5 shows the normalized IHI as a function of time after discharge for spent fuel from the GT-MHR. After about 100 years, the radiotoxicity is dominated by actinides, especially isotopes of plutonium. Figure 3-6 shows a comparison of the normalized IHI for spent fuel from the GT-MHR, a typical PWR, and the PC-MHR. The data for the PWR and PC-MHR were taken from Ref. 12. The curve for the PWR corresponds to a burnup of 33,000 Mw_t-d/MTHM. The curve for the PC-MHR represents deep burn of weapons-grade plutonium to an average burnup of about 590,000 Mw_t-d/MTHM, which corresponds to destruction levels of 65% of the total plutonium and 90% of the Pu-239. The lower radiotoxicity for the GT-MHR is a direct result of its lower plutonium content.

* For LWRs, the IHI is often normalized with respect to the MTHM associated with the spent fuel. As discussed in Section 3.1, the PWR requires nearly 5 times the heavy metal of the GT-MHR, on a per unit electrical energy basis. Hence, MTHM is not an appropriate normalization parameter when comparing the IHI for the two reactor types.

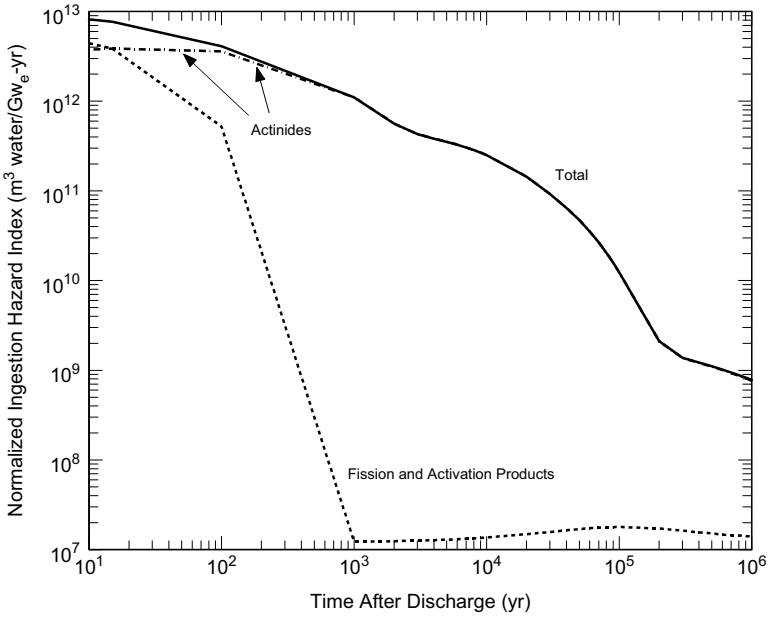

Figure 3-5. Normalized IHI for the GT-MHR

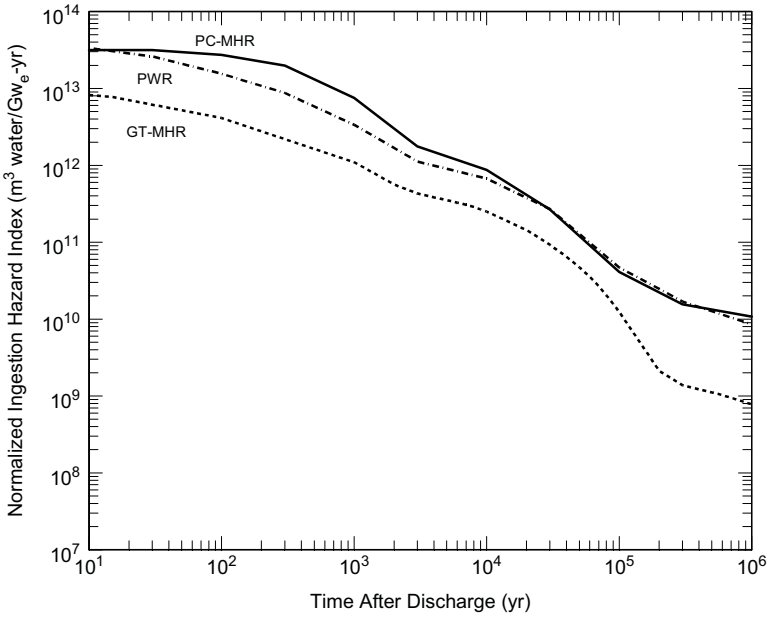

Figure 3-6. Comparison of the Normalized IHI for the GT-MHR, PC-MHR, and PWR

3.7 Decay Heat

The decay-heat load of spent fuel is an important consideration for designing waste packages and developing strategies for loading spent fuel into the repository. The decay-heat load is also a contributing factor to the thermal response and repository performance of the spent fuel, especially at earlier times when the decay-heat rates are higher. Figure 3-7 shows the decay-heat load of an average GT-MHR spent fuel element as a function of time after discharge from the reactor. The decay-heat load drops from about 16 w_t at 10 years after discharge to about 0.1 w_t at 10,000 years after discharge.

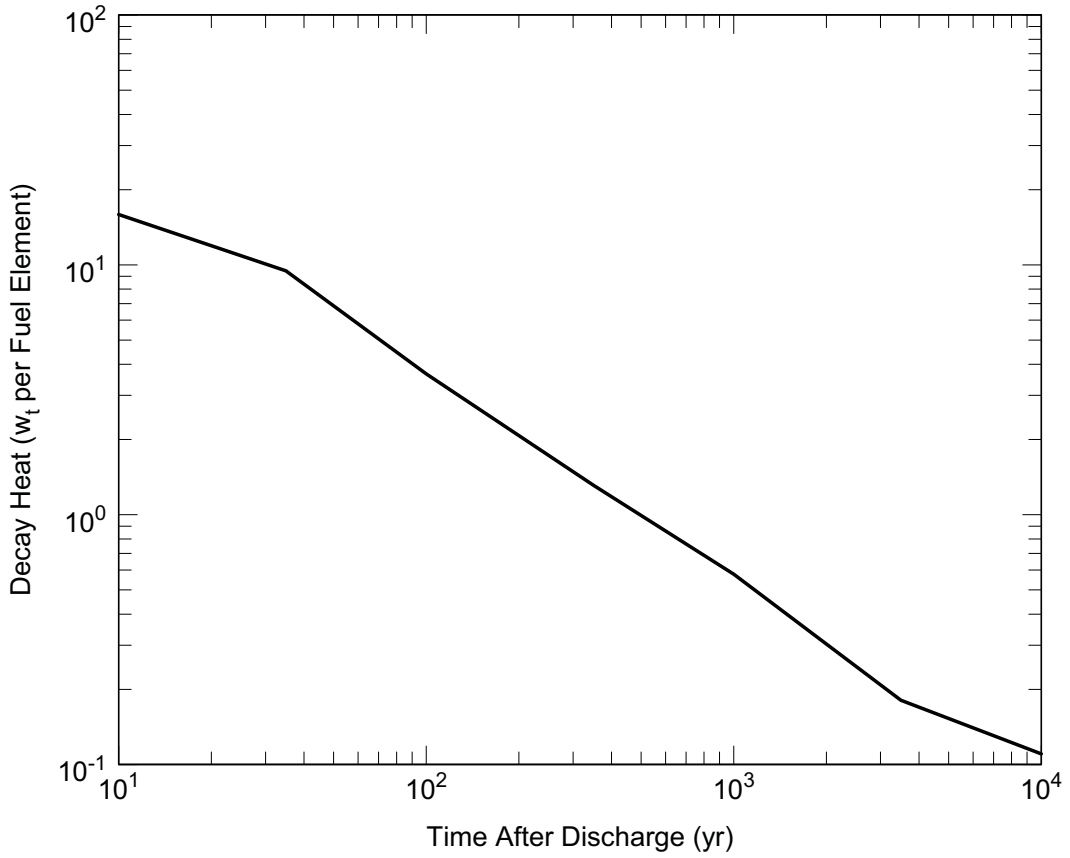

Figure 3-7. Decay-Heat Load of an Average GT-MHR Spent Fuel Element

Figure 3-8 shows a comparison of decay-heat loads for the GT-MHR, PC-MHR, and PWR, normalized with respect to electrical energy generated by the spent fuel. The curve for the PWR was taken from Ref. 29 and is representative of fuel irradiated to 33,000 Mw_t-d/MTHM. The decay heat load for PWR spent fuel is higher at earlier times because of its higher content of shorter-lived fission products. The decay heat load for PC-MHR spent fuel is higher at later times because of its higher actinide content. Because of its lower fission-product and transuranic inventories, the decay heat load of GT-MHR spent fuel is significantly lower than that for PWR spent fuel.

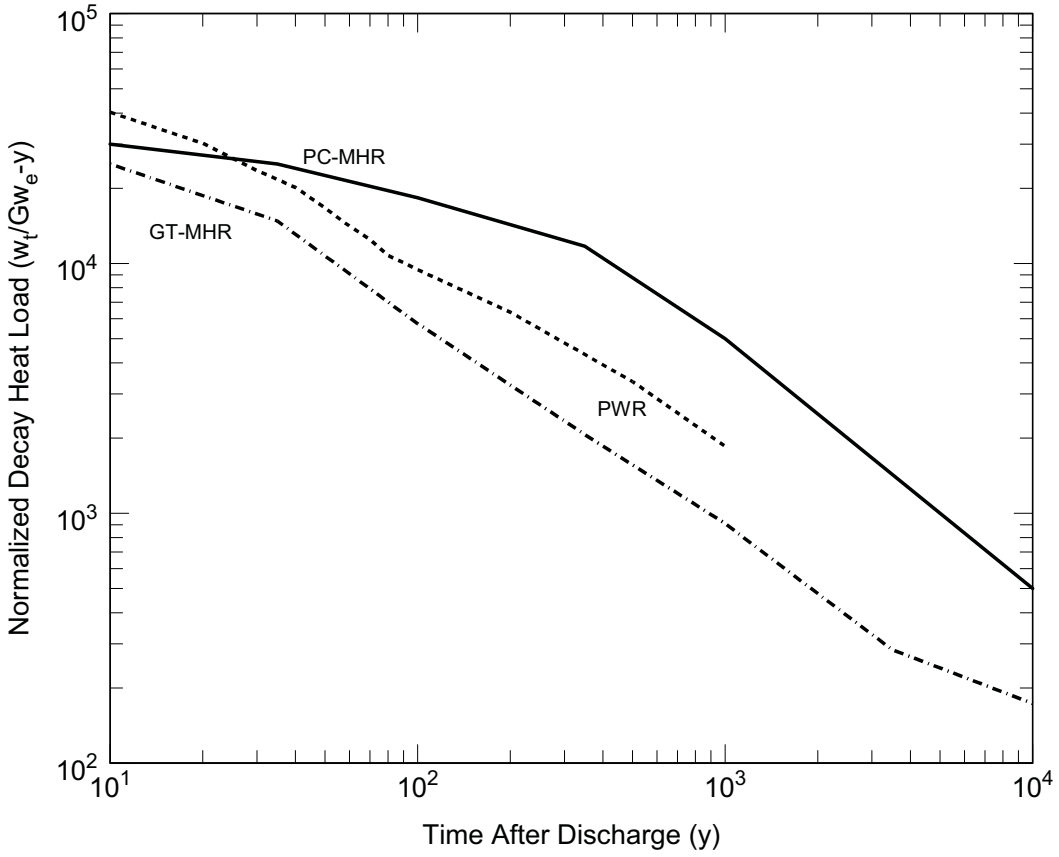

Figure 3-8. Comparison of Decay-Heat Loads for the GT-MHR, PC-MHR, and PWR

3.8 Proliferation Resistance

A highly desirable goal for any advanced reactor concept is to have a high degree of resistance to proliferation of weapons-usable materials throughout the fuel cycle, including after the spent fuel has been permanently disposed in a geologic repository. The proliferation resistance of the GT-MHR fuel cycle was assessed recently in response to requests from the TOPS* Task Force of the Nuclear Energy Research Advisory Committee (NERAC). As part of its work, the TOPS Task Force developed a systematic approach for assessing the proliferation resistance of a nuclear fuel cycle (Refs. 30 – 33). For all steps of the fuel cycle, ranging from mining of raw material to emplacement of spent fuel in the repository, the TOPS approach uses a ranking system to assess proliferation resistance with respect to material barriers, technical barriers, and institutional barriers. Based on this assessment it was concluded the GT-MHR fuel cycle was highly resistant to proliferation, particularly when compared to the LWR once-through fuel cycle. Key attributes of the GT-MHR and its fuel cycle that enhance proliferation resistance include the following:

- The GT-MHR uses low enriched fuel (< 19.9% U-235).
- The uranium is never in a metallic form.
- The GT-MHR operates at low-power density, with the fuel diluted within the relatively large volume occupied by the graphite fuel-element blocks. A large number of blocks would have to be diverted in order to obtain a sufficient quantity of heavy metal for potential weapons use.
- Because the fuel is irradiated to high burnup (~110,000 Mw_t/MTHM), the isotopic composition of the plutonium in discharged fuel is degraded and highly unfavorable for potential weapons use.
- The GT-MHR reactor core is refueled off-line, which greatly decreases the probability that low-burnup fuel could be diverted and processed to produce weapons-grade plutonium (WPu). In order to produce WPu, the GT-MHR would have to be refueled on a very frequent basis, which practically precludes its use for a weapons program.†
- As is the case for spent fuel discharged from any reactor, the radiation dose rate from GT-MHR spent fuel provides a radiological barrier that prevents access to the spent fuel for many decades, except in dedicated facilities that have remote-operations capabilities.
- It is significantly more difficult and expensive to recover heavy metal from coated-particle fuel than from metal-clad fuel.

* TOPS is an acronym for Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems.

† Another gas-cooled reactor design, the Pebble Bed Modular Reactor (PBMR), is refueled on-line, with the fuel-element pebbles passing through the core 10 to 12 times before discharge. With this on-line refueling scheme, it would be possible to discharge some pebbles prematurely to recover WPu, with only a modest impact on reactor operations and economics. In general, reactors that are refueled on-line are more reliant on institutional controls for resistance to proliferation.

- The spent fuel discharged from the GT-MHR is a robust final waste form that requires no additional processing for final disposal, other than interim storage and loading the spent fuel elements into a MPC.

An issue that has received increased attention in recent years is the potential requirement for long-term safeguarding of geologic repositories containing unprocessed spent fuel with significant quantities of plutonium (e.g., Refs. 34 and 35). Because the radiation barrier from spent fuel is initially dominated by shorter-lived fission products (e.g., 30.2-yr Cs-137), the dose rate from spent fuel decreases by several orders of magnitude after 300 to 400 years. With these much lower dose rates, it would not be especially difficult to chemically process the spent fuel to recover plutonium. Peterson has concluded that recovery of canisters from a repository would also not be especially difficult or expensive, and the potential plutonium production rates from mining a repository would greatly exceed that from using dedicated reactors and processing facilities (Ref. 34). As discussed in Section 2.5, neither the EPA nor NRC requires evaluation of this "plutonium mine" scenario for assessing repository performance, and the DOE has chosen not to evaluate this scenario in its Yucca Mountain performance assessments (Ref. 1).

In general, high-quality, WPu is preferred for use in weapons because it has a lower heat load and lower neutron generation rate from spontaneous fission than reactor-grade plutonium (RPu), on a per unit mass basis. A weapon can be constructed using RPu, but the higher heat load adds complexities to the design and manufacture of the weapon, and the higher neutron generation rate translates into less predictability of the yield and a higher probability that the weapon will detonate with lower than maximum yield. The heat load is produced primarily by the Pu-238 and Pu-241 decay chains. After a few hundred years, the heat loads of WPu and RPu are about the same, because of the decay of these shorter-lived isotopes (Ref. 34). The decay process also shifts the isotopic composition of the spent fuel to somewhat higher levels of Pu-239. This aging process clearly increases the attractiveness of plutonium for weapons use.

Table 3-9 gives the quantity of total plutonium and Pu-239 in commercial spent nuclear fuel (CSNF) waste packages and in 42-element MPCs for GT-MHR and PC-MHR spent fuel. The information for the CSNF waste packages was obtained from Refs. 1 and 36. All of the waste packages described in Table 3-9 are of similar size. The plutonium contents given in Table 3-9 assume several hundred years of radioactive decay and neglect the small contributions that 87.7-yr Pu-238 and 14.4-yr Pu-241 would provide at that time. Using the CSNF values given in Table 3-9, the Yucca Mountain repository would contain about 550 metric tons of weapons-usable plutonium, which is sufficient to make on the order of 50,000 nuclear weapons. Most of the individual CSNF waste packages would contain sufficient plutonium to make several weapons. For example, assuming 10 kg of RPu is required to make a weapon, the 21-PWR Absorber Plate package would contain sufficient plutonium for 8 weapons. Some of the CSNF waste packages also have more favorable plutonium isotopics, with up to 79% Pu-239 after several hundred years of decay. For GT-MHR and PC-MHR spent fuel, the plutonium is diluted into a much larger volume and the plutonium isotopics are more degraded than those for CSNF. A potential proliferator would have to retrieve several GT-MHR waste packages in order to obtain sufficient plutonium for a single weapon, and the plutonium would be difficult to extract from the coated-particle fuel.

Table 3-9. Plutonium Content in Waste Packages[a]

Package Design	Description	Number of Packages Slated for Yucca Mountain	Average Burnup per Assembly or Element (Gw_t-d/MTHM)	Average MTHM per Assembly or Element	Plutonium Content per Waste Package (kg)	Pu-239 Content (%)
21-PWR Absorber Plate	Holds 21 PWR assemblies with absorber plates for criticality control	4500	41.5	0.43	83.2	63
21-PWR Control Rod	Holds 21 higher reactivity PWR assemblies with control rods for criticality control	100	19.6	0.368	50.4	77
12-PWR Long	Holds 12 PWR assemblies that are longer than typical assemblies	170	46.3	0.54	60.2	63
44-BWR	Holds 44 BWR assemblies	3000	34.1	0.177	53.5	60
24-BWR	Holds 24 higher reactivity BWR assemblies	90	8.1	0.167	13.2	79
GT-MHR MPC	Holds 42 GT-MHR fuel elements	—	110	4.425×10^{-3}	2.7	55[b]
PC-MHR MPC	Holds 42 PC-MHR fuel elements	—	590	7.7×10^{-4}	7.5	43

Note
a. The plutonium contents neglect 87.7-yr Pu-238 and 14.4 yr Pu-241 and are representative of RPu after several hundred years of decay.
b. The isotopic distribution of plutonium in fissile and fertile fuel is nearly the same.

4. GT-MHR Multipurpose Canister

As discussed in Section 1.3, an MPC design has been developed through the preliminary design stage for PC-MHR spent fuel. This MPC design is described in detail in Ref. 10. With little or no modification, this MPC design could also be used for storage, transportation, and disposal of GT-MHR spent fuel.

4.1 Design Requirements

In general, the PC-MHR MPC was designed to be consistent with existing requirements for CSNF canisters, and to interface seamlessly with the existing and planned infrastructure for transportation and disposal of CSNF. Design requirements are given in Table 4-1.

Table 4-1. MPC Design Requirements

Design Attribute	Requirements
Weight	Loaded MPC < 55 tons
	Transportation Cask < 50 tons
	Lifting Yoke < 10 tons
	Lifting Adapter < 10 tons
	Total Weight < 125 tons
Dimensions	Outer Diameter < 63 in.
	Length < 193 in.
Materials	300 Series Stainless Steel
Service Life	Pre-Disposal Storage: 100 yr
	Post-Disposal Retrievability: 100 yr
Criticality	k_{eff} (multiplication factor) < 0.95[a]
Shielding[b]	Top of MPC Shield Plug: < 200 mrem/hr
	Sides of Storage Cask: < 20 mrem/hr
	Top and Bottom of Storage Cask: < 50 mrem/hr
Thermal	Fuel (storage and transportation) < 427°C
	Fuel (repository) < 232°C
	MPC Metal (storage) < 371°C
	MPC Metal (transport) <149°C
	Concrete Storage Cask < 177°C
Closure	Closures Must Be Welded
	Must allow for opening of MPC to recover spent fuel
Interfaces	Concrete Storage Cask
	Shipping Cask
	Repository Overpacks
Structural	Must comply with 10CFR71, “Packaging and Transportation of Radioactive Material,” including requirements for drop tests, puncture tests, thermal tests, water-immersion tests, and crush tests.

Notes
a. The calculated value for k_{eff} accounts for burnup of the spent fuel and an allowance of 0.03 to account for calculational uncertainties and bias.
b. These requirements are for 1-yr old spent fuel stored in a concrete storage cask.

4.2 Design Description

During the PC-MHR MPC preliminary design phase, detailed thermal, structural, criticality, shielding, cost, and fabrication evaluations were performed for five different MPC designs (Ref. 10). The reference design, called the "channel-basket" design, was selected because it satisfied all regulatory and design requirements with lower cost and relative ease of fabrication. Figure 4-1 shows cross-section end elevation views of the channel-basket MPC.

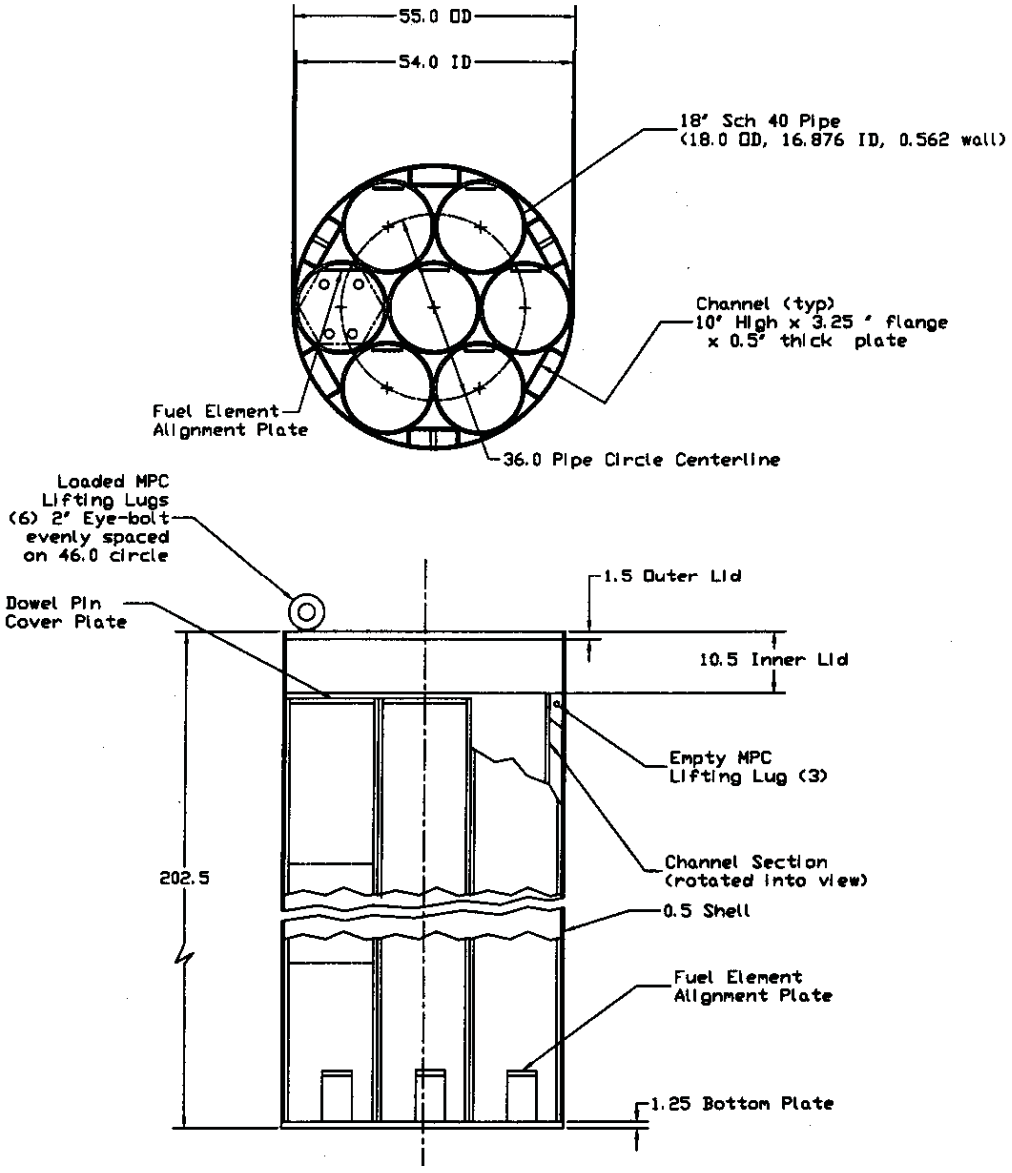

Figure 4-1. Channel-Basket MPC (dimensions are in inches)

The design consists of a cylindrical shell structure with an interior array that consists of a cluster of closely packed pipes and channels to form the so-called channel basket, which houses 42 PC-MHR or GT-MHR spent fuel elements, with 6 elements in each of the 7 pipes. The pipes are 18-in. Schedule 40 and are welded together during assembly. Channels are welded to the periphery of the pipe structure to form a rigid frame. The shell of the MPC has an outer diameter of 55 in. with a wall thickness of 0.5 in. The inner and outer lids are attached with full-penetration welds after the MPC is loaded with spent fuel. The structural shell and lid material is 304L stainless steel. The overall length of the MPC is 202 in. For comparison, a PWR canister that holds 21 PWR spent fuel assemblies has similar overall dimensions, with an outer diameter of 64.7 in. and a length of 203.3 in. (Ref. 1).

The structural, thermal, and shielding requirements for an MPC containing GT-MHR spent fuel are nearly identical to those for the PC-MHR MPC. However, criticality evaluations of an MPC containing fuel discharged from a uranium-fueled GT-MHR could be significantly different from those for a plutonium-fueled PC-MHR. For the PC-MHR MPC, the criticality analysis was performed using the MCNP computer code, assuming the MPC contained 1-yr old spent fuel and was flooded with water while the MPC was in a concrete storage cask (Ref. 10). The calculation was based on conservative assumptions, including complete burnout of the burnable poison (Er-167), neglect of fission-product poisoning, and use of peak-fuel temperatures. The calculated multiplication factor (k_{eff}) was 0.54, which includes an allowance of 0.03 required by the NRC for calculational uncertainties and bias.

A detailed criticality analysis has not been performed for the GT-MHR MPC, but given the conservative assumptions used for the PC-MHR MPC, the criticality potential for the GT-MHR MPC can be assessed by comparing its content of fissile material to that of the PC-MHR MPC. As indicated in Table 4-2, the GT-MHR MPC would contain about 30% more fissile material by weight than the PC-MHR MPC, because of the U-235 remaining in GT-MHR spent fuel. However, if the fissile-material content is weighted according to the thermal fission cross sections of the respective nuclides, the weighted total for the GT-MHR MPC is somewhat less than that of the PC-MHR MPC. This latter comparison is equivalent to comparing overall macroscopic cross sections, and is likely to be more relevant for assessing criticality potential. Based on this assessment and the large design margins in k_{eff} calculated for the PC-MHR MPC, an MPC loaded with GT-MHR spent fuel should also have a large design margin in k_{eff}, but more detailed calculations should be performed for confirmation.

Table 4-2. MPC Fissile Material Content[a]

Nuclide	Thermal Fission Cross Section (b)	Content per MPC (kg)	
		GT-MHR	PC-MHR
U-235	585	9.02	—
Pu-239	750	1.46	3.54
Pu-241	1010	0.62	5.08
	Total (kg)	11.10	8.62
	Weighted Total (kg-b)[b]	7,000	7,790

Notes
a. Values are for 1-yr old spent fuel.
b. This quantity is the sum of the products, content (kg) and thermal fission cross section (b).

Table 4-3 provides a comparison of the GT-MHR MPC with the 21-PWR Absorber Plate waste package (see Fig. 4-2). As discussed previously, the two canisters have nearly identical overall dimensions. Because of its low power density, the GT-MHR requires about 12 times the number of waste packages required by the PWR, on a per unit electrical energy basis. However, as indicated by the data in the table, the thermal and structural loads are much higher for the PWR waste package. Also, because of its higher loading of fissile material, neutron poison must also be added to the PWR waste package to satisfy criticality-control requirements. The PWR waste package includes a corrosion barrier composed of nickel-based Alloy 22 and a titanium drip shield manufactured from Alloy 22 and titanium. This drip shield provides an additional barrier to corrosion and protects the waste package from rock fall.* These additional barriers are not required for GT-MHR spent fuel because of the long-term containment provided by the TRISO coatings (see Section 6) and the use of backfill to protect against rock fall.

The greater demands on the PWR waste-package design translate into much higher unit costs. On a per unit electrical energy basis, the overall waste-package costs may be somewhat higher for the GT-MHR, but these costs should account for only about 25% of the total life-cycle cost of the repository, based on the independent cost estimate described in Ref. 37. Also, as discussed in Section 3.8, the relatively large number of waste packages required for the GT-MHR provides a significant benefit in terms of resistance to proliferation.

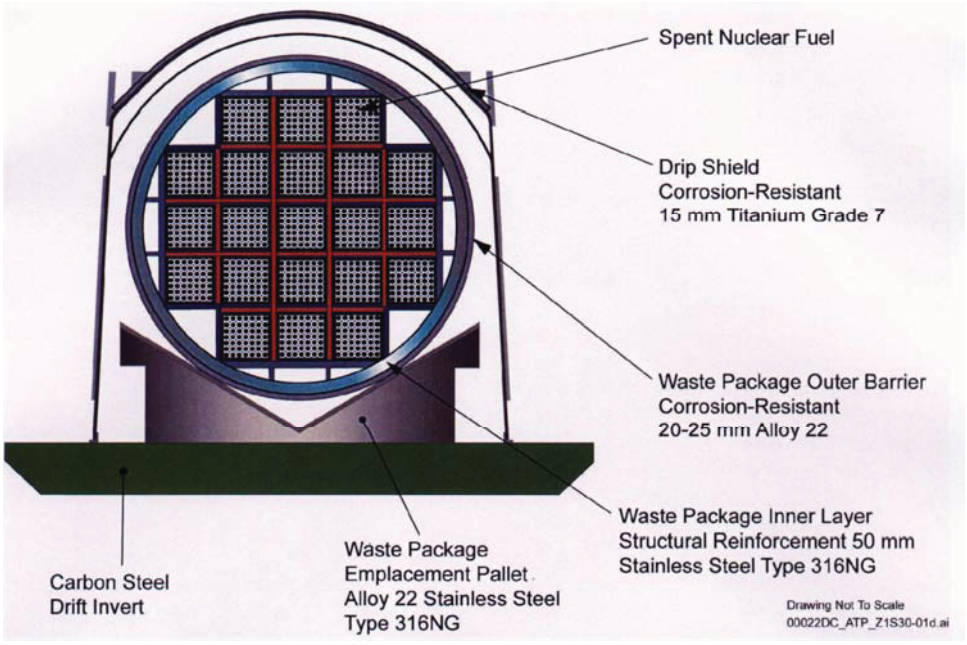

Figure 4-2. Commercial PWR Waste Package (reproduced from Ref. 1)

* As indicated in Table 4-3, the drip shield adds significant costs to the LWR waste package. Independent reviewers have questioned the added value of this drip-shield design, and have recommended that a value-engineering study be performed to determine if it provides a cost-effective enhancement (Ref. 36).

Table 4-3. Comparison of GT-MHR MPC with 21-PWR Absorber Plate Waste Package

	GT-MHR MPC	**21-PWR Absorber Plate[a]**
Capacity	42 Elements	21 Assemblies
Structural Shell Material	304L Stainless Steel	316 NG Stainless Steel
Structural Shell Thickness (cm)	1.27	5
Corrosion Barrier Material[b]	—	Alloy 22
Corrosion Barrier Thickness (cm)	—	2 – 2.5
Drip Shield Material	—	Alloy 22 and Titanium
Weight Without Fuel (mt)	12.8	26
Weight With Fuel (mt)	18.5	42.3
Outside Diameter (m)	1.397	1.644
Length (m)	5.144	5.165
Decay Heat Load (kw_t)	0.669[c]	11.53
Plutonium Content (kg)	3.13[c]	96.0
Pu-239 Content (kg)	1.46 (47%)[c]	52.8 (55%)
Uranium Content (kg)	160.5[c]	8570
U-235 Content (kg)	9.0 (5.6%)[c]	72.2 (0.84%)
Neutron Poison Required?	no	yes
Canisters Required Per Gw_e-yr	37	3
N^{th} Unit Cost (Year 2000 $K)	87.2[d]	390 (waste package)
		313 (drip shield)
		56 (pallet)
		759 (total)

Notes
a. Design data are representative of an average canister and are taken from Refs. 1 and 36. Cost data are taken from Ref. 37.
b. This material surrounds the 21-PWR Absorber Plate canister.
c. Values are for 10-yr old spent fuel.
d. Derived from the Ref. 10 estimate of $75.2K (in 1994 $) and escalated to year 2000 $ using the escalation factors given in Ref. 38. There are some additional costs for a railcar (which functions as a pallet) and backfill, but these costs were not estimated.

5. Spent Fuel Handling and Repository Loading

Remote-handling systems are used to remove spent-fuel elements from the GT-MHR core and transfer them to dry storage wells that are immersed in a pool of cooling water. These cooled storage wells are located adjacent to the reactor module. After 1 year of cooling in the storage wells, the spent-fuel elements are transferred to an MPC, which can be housed within a passively-cooled concrete storage cask (see Fig 5-1) or vault for on-site storage. After 5 to 10 years of on-site storage, the loaded MPCs are transferred to NRC-licensed transportation casks for shipment via railcar to the repository (see Figs. 5-2 and 5-3).

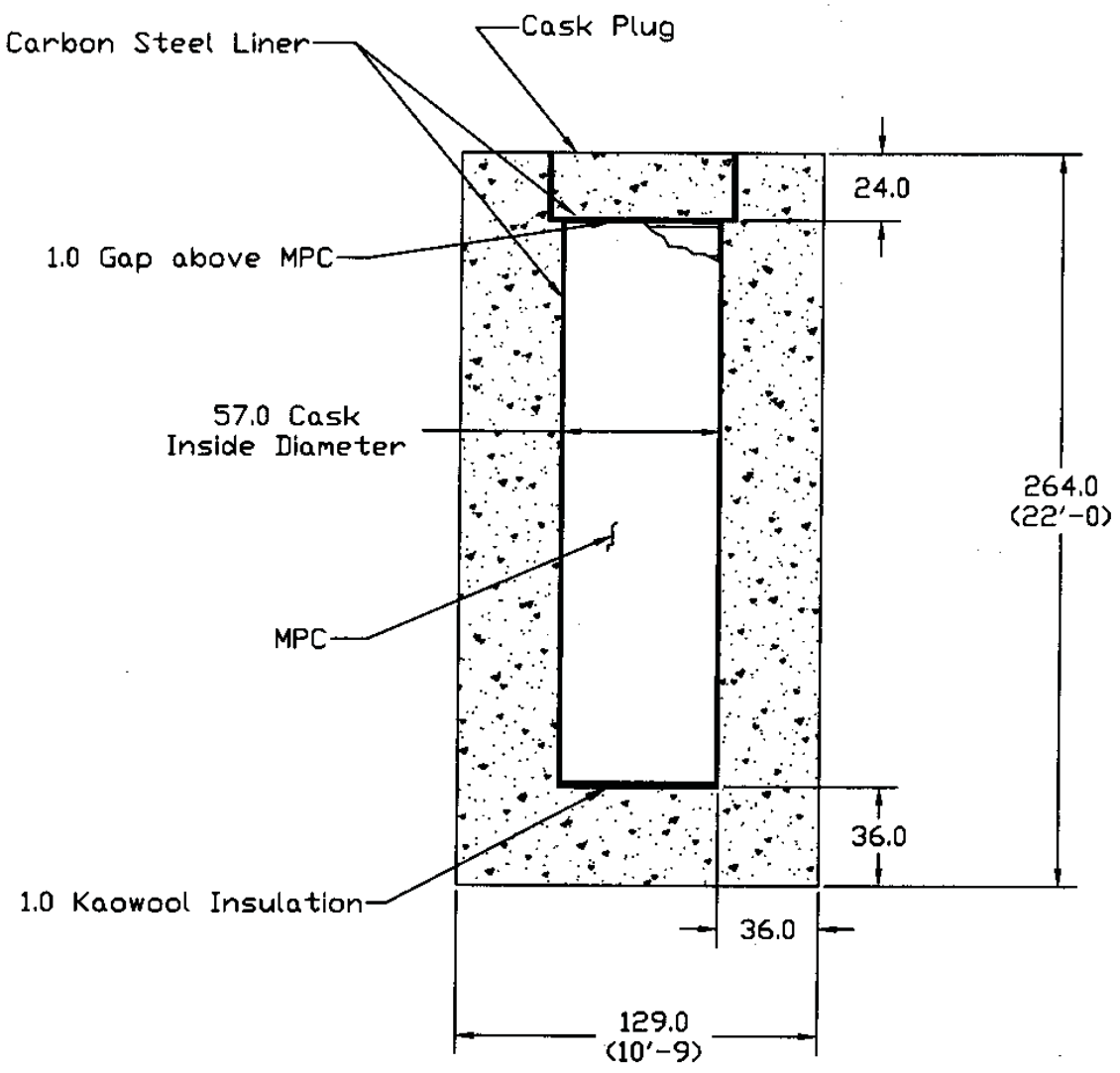

Figure 5-1. Concrete Storage Cask (dimensions are in inches)

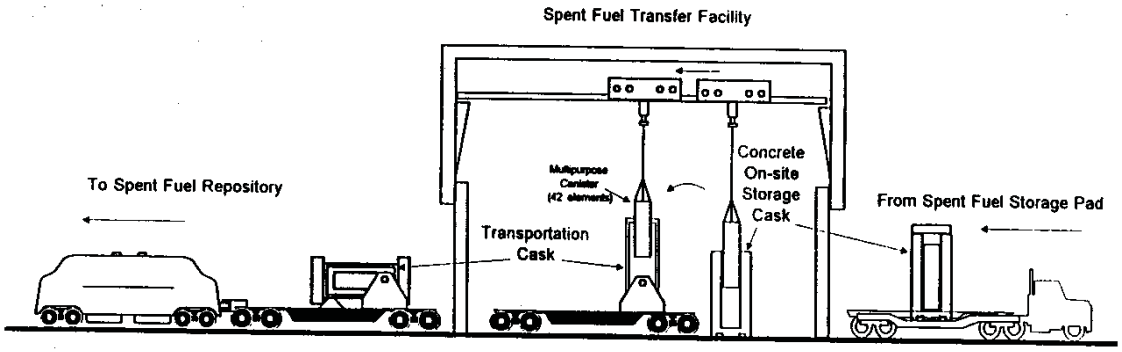

Figure 5-2. Transfer of MPC from Storage Cask to Transportation Cask

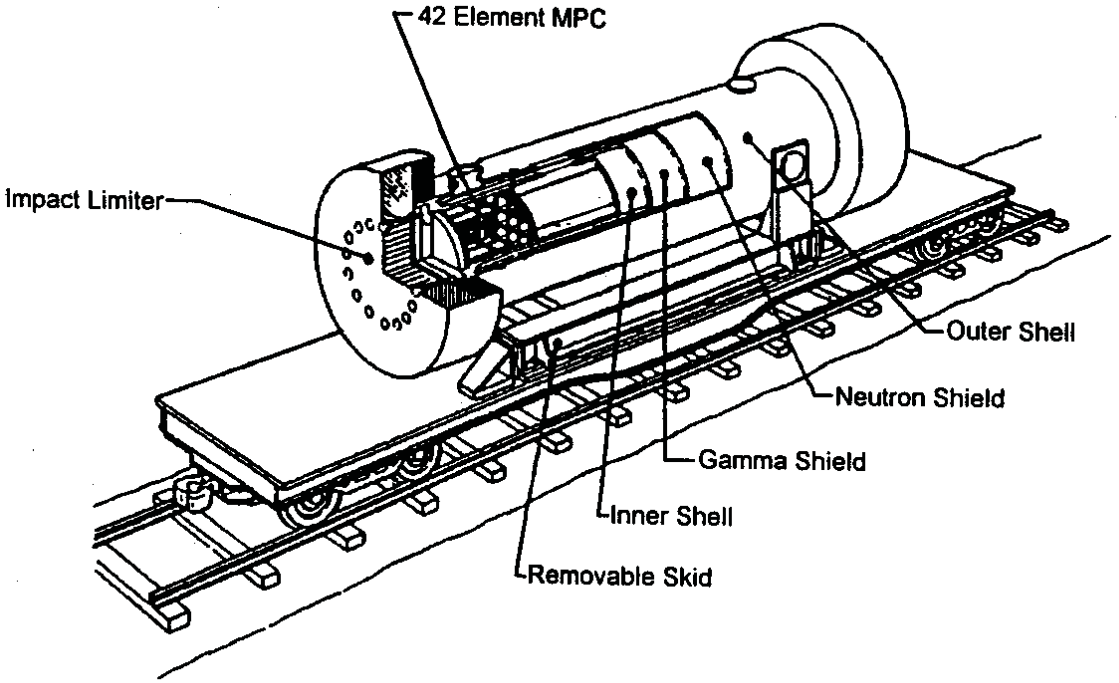

Figure 5-3. Transportation Cask

Repository designs and loading strategies for disposal of PC-MHR MPCs are described in detail in Ref. 11. Design of the repository is determined by a number of factors, including thermal loading and maintaining structural integrity of the repository after the drift tunnels are excavated and loaded with canisters. The strategies for disposal of PC-MHR spent fuel were developed based on the assumption that spent fuel is loaded into the repository at 5 to 10 years after discharge. Over this time frame, the decay-heat load for GT-MHR spent fuel is only somewhat lower than that for PC-MHR spent fuel (see Fig 3-8). In terms of factors that affect repository design, there are only small differences between GT-MHR and PC-MHR spent fuel, and the strategies described in Ref. 11 are directly applicable to disposal of GT-MHR MPCs.

At the time the Ref. 11 report was prepared (1995), DOE was considering three thermal limits for disposal of CSNF and defense waste at the Yucca Mountain repository:

- A “cold” repository with a thermal loading limit of ~20 kw_t/acre.
- An “intermediate” repository with a thermal loading limit of ~57 kw_t/acre.
- A “hot” repository with a thermal loading limit of ~100 kw_t/acre.

The cold repository results in lower canister and fuel temperatures, and has the least impact on the natural geologic barriers provided by the repository, but it requires more land. The hot repository allows for a denser loading of canisters, but may require active ventilation to satisfy thermal limits for the waste package, fuel, and repository drift walls. The intermediate repository is a compromise between the hot and cold designs.

Figure 5-4 shows the baseline repository layout for PC-MHR MPCs on a per unit acre basis, with the area defined as a 64 m × 64 m square. This configuration allows for seven 4.3-m diameter tunnels, with a spacing between tunnel centerlines of 8.6 m. The ratio of tunnel diameter to tunnel-to-tunnel centerline spacing is called the extraction ratio, which is equal to 0.5 for this design. Because of its low decay-heat load, an extraction ratio of 0.5 was determined to be feasible for disposal of PC-MHR spent fuel. This design allows for 7 drift tunnels over a 64-m width and 11 MPCs per tunnel over a 64-m length, for a total of 77 MPCs on a square-shaped acre. Because of repository structural considerations, a denser loading is not possible for PC-MHR or GT-MHR spent fuel. For GT-MHR spent fuel, the heat load per unit area would be 51.5 kw_t/acre for 10-yr old spent fuel and 78.6 kw_t/acre for 5-yr old spent fuel.

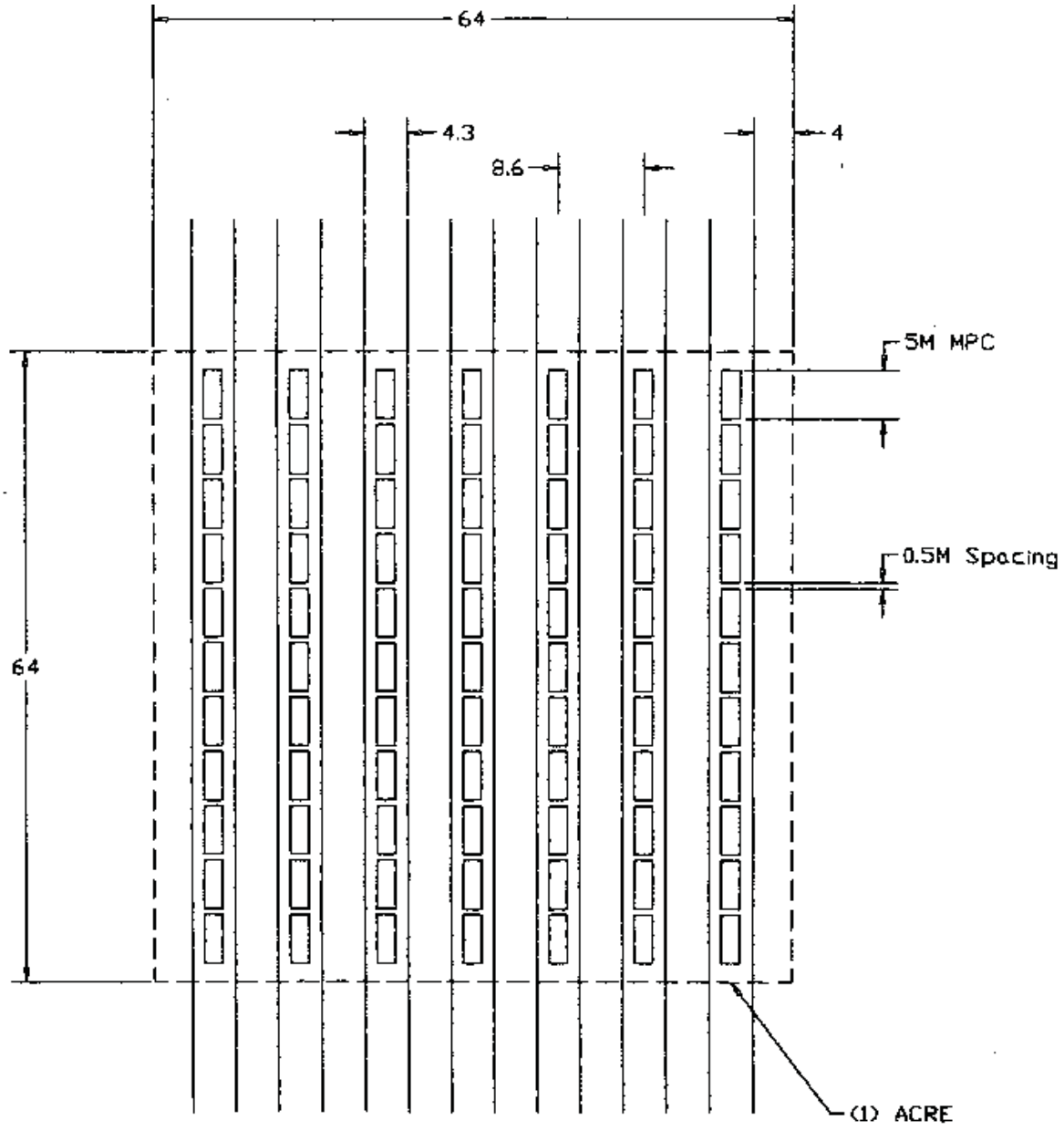

Fig. 5-4. Repository Layout for Disposal of GT-MHR MPCs (dimensions are in meters)

In Ref. 11, a comparison of repository parameters was made for disposal of PC-MHR and typical PWR spent fuel. Results of this comparison are summarized in Table 5-1 for the intermediate repository design loaded with 10-yr old spent fuel and no active cooling of the drift tunnels. Based on these results, PC-MHR spent fuel would require about 1/2 the repository area, about 2/3 of the tunnel volume, and about twice the tunnel length of PWR spent fuel, on a per unit electrical energy basis. Because of the high decay-heat load of PWR spent fuel, only four canisters can be loaded per acre.

Table 5-1. Repository Parameters for the Intermediate Repository Design (57kw_t/acre)

	PC-MHR	PWR
Decay-Heat per Canister (kw_t)	0.756[a]	13.2[b]
Required Repository Area (acres/ Gw_e-yr)	0.38	0.74
Required Tunnel Volume (1000 m^3/ Gw_e-yr)	2.47[c]	3.65[d]
Required Tunnel Length (1000 m/ Gw_e-yr)	170[c]	95[d]
Drift Wall Temperature (ºC)	180	105
Fuel/Clad Temperature (ºC)	190	320
Canisters per Acre	77	4

Notes
a. 42-element MPC.
b. 21-assembly canister.
c. Based on a tunnel diameter of 4.3 m and an extraction ratio of 0.5.
d. Based on a tunnel diameter of 7 m and an extraction ratio of 0.3.

As discussed in Ref. 1, the DOE has changed its repository design and loading strategy significantly since 1995. For the current design, the canisters are placed end-to-end (10 cm apart) in 5.5-m diameter drift tunnels, with a center-to-center spacing between tunnels of 81 m, which corresponds to an extraction ratio of 0.068. With this low extraction ratio, the boiling-point isotherms generated from individual drifts do not overlap, which creates cooler zones between the drifts that remain below the boiling point. During earlier time periods when the spent fuel is still relatively hot, water that percolates to the repository will tend to migrate to these cooler zones and avoid the waste packages. With this design, DOE has estimated that approximately 1150 acres would be required for disposal of the spent fuel and defense waste associated with the Yucca Mountain statutory limit of 70,000 MTHM. If all of the 70,000 MTHM came from PWRs with an average loading of 0.43 MTHM per assembly, then the repository would contain about 7,750 21-assembly waste packages, which corresponds to a loading of about 6.75 waste packages per acre. Assuming a decay-heat load of 11.53 kw_t per waste package, the repository heat load would be about 78 kw_t/acre. With end-to-end canister spacing, the average linear heat load in the repository is about 1.42 kw_t/m (Ref. 1).

The earlier Yucca Mountain repository designs included adding backfill to the drift tunnels after they were loaded with canisters. The backfill provides protection against rock fall and provides an additional engineered barrier to transport of water to the waste package and transport of radionuclides from the waste package, particularly if a highly sorbing material is used for the backfill. For the current design, backfill has been eliminated and an active air cooling system has been added in order to lower waste-package and fuel temperatures, and to

ensure that the boiling-point isotherms from adjacent drifts do not coalesce. For the current design, forced-air cooling is required for 50 years after emplacement in order to meet the design requirements (Ref. 1). This cooling system removes 70% of the decay-heat load, which requires six large exhaust fans and about 7.5 Mw_e of continuous power. The volumetric airflow rate in an emplacement drift is approximately 15 m^3/s.

Because of its low decay-heat load, high-temperature capability, and high resistance to corrosion, GT-MHR spent fuel requires a less sophisticated and less expensive waste package design than that for CSNF (see Section 4). For these same reasons, it is expected that a less sophisticated and less expensive repository design would also be required for disposal of GT-MHR spent fuel. The repository would be loaded with GT-MHR MPCs according to the layout shown in Fig. 5-4. The MPCs are loaded into the emplacement drifts on railcars, after which the railcars are locked in place to serve as pallets. After a prescribed period of monitoring and surveillance, the drifts are backfilled with a low-permeability, highly-sorbing material (e.g., bentonite).

A thermal model of a repository loaded with PC-MHR MPCs is described in Ref. 6. This model was used to calculate temperatures of the fuel, canister, and repository drift walls as function of time after emplacement. It was assumed there were no convective heat-removal mechanisms. Figure 5-5 shows the predicted drift-wall and centerline fuel/graphite temperatures. Because of the low power density, temperature gradients from the centerline to the outer wall of the canister are small, and the centerline temperature is a reasonable and somewhat conservative estimate of temperatures at other locations within the waste package. The centerline fuel/graphite temperature reaches a peak value of 214ºC at about 500 years. The temperature then drops with time as the decay heat decreases and heat rejection to the ultimate heat sink increases. Also shown on Fig. 5-2 is the boiling temperature at the repository elevation (96ºC for Yucca Mountain). For about 3,000 years, the drift wall temperature remains above the boiling point, and the waste packages should remain dry for this time period. Because of the lower decay-heat load of GT-MHR spent fuel (see Fig. 3-7), temperatures would be lower for a repository loaded with GT-MHR spent fuel, which would allow water to contact the waste package at an earlier time. Of course, given the very long time periods over which the calculations are performed, predictions of the thermal behavior of the waste packages and repository are highly uncertain. In general, it is probably reasonable to assume the temperature response of the waste packages is relatively slow with time and that the waste packages eventually cool to ambient temperatures.

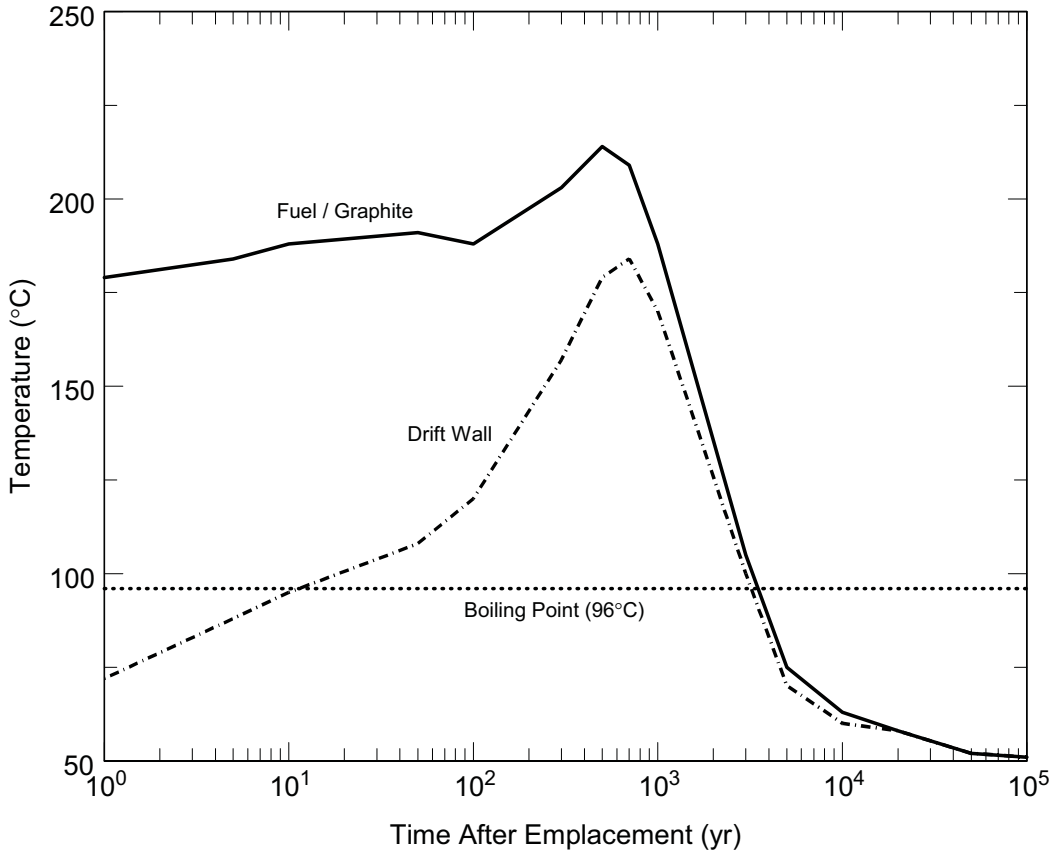

Figure 5-5. Repository and Fuel/Graphite Temperatures After Emplacement of PC-MHR Spent Fuel

6. Performance in a Geologic Repository

As described in Ref. 1, the DOE has performed a comprehensive, probabilistic assessment of the long-term radiological consequences resulting from disposal of CSNF and defense high-level waste in the Yucca Mountain repository. In order to judge regulatory compliance of the repository, the DOE has assumed the proposed EPA and NRC regulations described in Section 2 are applicable. Dose rates to the RMEI were calculated for the nominal, volcanic, and human-intrusion scenarios, and radionuclide concentrations in the groundwater were calculated to show compliance with the proposed EPA regulations described in Section 2.4. The performance assessments show the repository should pose a negligibly small radiological risk for time periods well beyond the proposed 10,000-yr compliance period, primarily because of the multiple engineered barriers provided by the waste packages and the natural barriers provided by the remote, arid Yucca Mountain site.

Although an assessment of the magnitude described in Ref. 1 is well beyond the scope of the present study, performance of GT-MHR spent fuel in a geologic repository can be assessed by comparing the characteristics of GT-MHR spent fuel with those of CSNF, and then using engineering judgement to determine how the differences would affect repository performance. Assuming performance of all of the natural and engineered barriers were identical, GT-MHR spent fuel may offer some performance benefits because of its lower radionuclide inventory, with the possible exception of radiological consequences caused by C-14 release (see Section 3.5). The engineered barriers for CSNF and GT-MHR spent fuel and waste packages differ in several important ways:

- For CSNF, the spent fuel itself does not provide a significant barrier to corrosion and release, and a heavy reliance is placed on the waste package for long-term protection. The engineered barriers include a thick, 5-cm structural shell composed of 316 NG stainless steel, an additional 2 to 2.5 cm of Alloy 22, and a drip shield composed of Alloy 22 and titanium.

- For GT-MHR spent fuel, the structural shell is composed of corrosion-resistant 304L stainless steel, but it is only 1.27-cm thick. The drift tunnels will be backfilled with a low-permeability, highly-sorbing overpack, which should provide some additional protection from groundwater attack. Given the expensive, robust design of the CSNF waste package, it is reasonable to conclude that GT-MHR MPCs will fail at a higher rate. Once an MPC has failed, the fuel elements will be exposed to groundwater. However, because of the ceramic nature of the graphite fuel elements and fuel-particle coatings, the corrosion rates are expected to be very low, and transport of any radionuclides that are released will be greatly impeded by the overpack.

The long-term radiological consequences of a repository containing GT-MHR spent fuel will be largely dependent on long-term performance of the coated particles and the corrosion/leach rates of the graphite fuel elements. Several researchers have concluded that graphite, pyrocarbon, and SiC are all excellent materials for isolation of high-level waste in a geologic repository. The expected performance of these engineered barriers for GT-MHR spent fuel is discussed in the following sections.

6.1 Corrosion Rates of GT-MHR Spent Fuel Materials

As part of a DOE-sponsored program in the early 1980s, ORNL developed coated-particle waste forms, based on HTGR coated-particle fuel technology (Ref. 39). Sol-gel technology with internal gelation was used to manufacture crystalline microspheres of simulated nuclear waste. The microspheres were coated with pyrocarbon and different combinations of pyrocarbon and SiC in a fluidized-bed coater. Leach tests were performed on the coated waste particles and more conventional glassified waste forms (e.g., borosilicate glass). Coated waste particles (including those coated with pyrocarbon only) leached at rates slower than could be detected by sensitive analytical techniques, including atomic absorption and inductively coupled plasma atomic emission. In contrast, radionuclide release from the glassified waste forms was readily detected.

Also during the early 1980s, the corrosion behavior of graphite, glassy carbons, pyrocarbon, and SiC was investigated by Pacific Northwest Laboratory (PNL), for the purpose of evaluating improved barriers for nuclear waste isolation (Refs. 40 and 41). In order to obtain meaningful corrosion data in a reasonable time, tests were performed using deionized water at elevated temperatures (200 to 300°C) in a pressurized autoclave. Measurements of CO and CO_2 were used to quantify the corrosion rates. The test materials consisted of powdered graphite, glassy carbon powders, and supercalcine and ZrO_2 beads (~500 μm in diameter) coated with layers of pyrocarbon and both pyrocarbon and SiC. For some of the graphite corrosion tests, a Co-60 gamma source was used to determine if the corrosion rates were enhanced by radiolysis. For the coated beads, no information about the coating properties (thickness, deposition temperatures, as-manufactured quality, etc.) was reported. The following results were obtained:

- At 250°C, the leach rate of graphite in deionized water was more than a factor of 10^5 slower than measured leach rates for waste glass.
- The corrosion rates of pyrocarbon and SiC were not significantly different, but were 5 to 10 times higher than the graphite corrosion rates.
- The corrosion rates of glassy carbon heat treated at 2600°C (GC-2600) and graphite were approximately the same. However, the corrosion rate of glassy carbon heat treated at 1000°C (GC-1000) was several times higher.
- Radiolysis had little effect on the graphite corrosion rate.

In Ref. 41, the author acknowledges that the small difference in the pyrocarbon and SiC corrosion rates may be the result of poor-quality, porous SiC, i.e., the corrosion attributed to the SiC may actually be that of underlying pyrocarbon. Additional data on well characterized materials are needed to resolve this discrepancy.

The data reported in Ref. 41 for two pyrolytic carbons and graphite were fitted to Arrhenius functions of temperature. These materials are described in Table 6-1, which also gives the fitted values for the pre-exponential and activation temperature. Figure 6-1 shows the corrosion rates as a function of inverse absolute temperature. The activation energies for graphite and PyC-2 are nearly the same and are about 50% lower than that for PyC-1. When

extrapolated to lower temperatures, the corrosion rates for all three materials are very low, with PyC-1 showing the lowest extrapolated rate because of its higher activation energy.

Table 6-1. Corrosion-Rate Parameters for Pyrocarbon and Graphite

Material	BET Surface Area (cm^2/g)	Form	Corrosion Rate = A_0 Exp(-T_A/T) (g/cm^2-d)	
			A_0	T_A
PyC-1	360	Coated Particle	16	9,460
PyC-2	190	Coated Particle	0.12	6,490
Graphite	12,000	Powder	0.0045	6,240

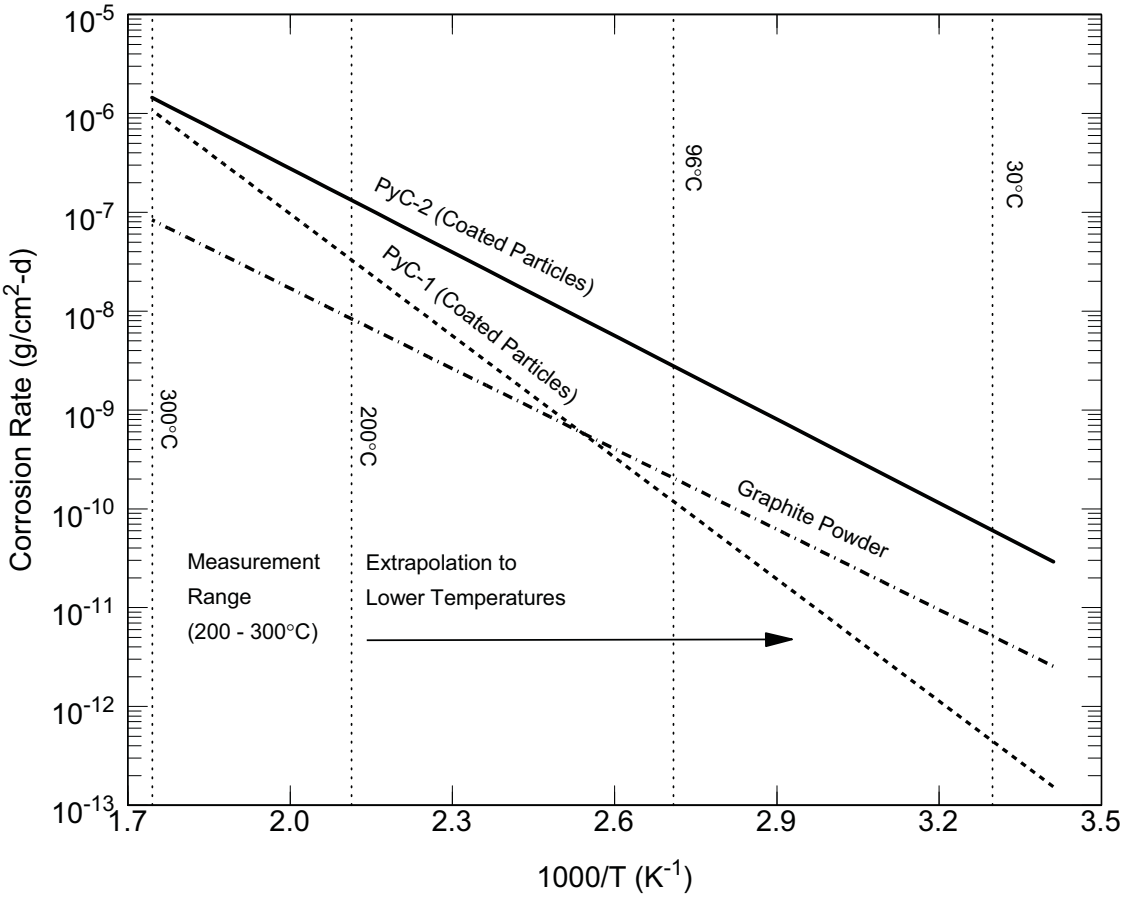

Figure 6-1. Corrosion Rates for Pyrocarbon and Graphite

The curves shown on Fig. 6-1 represent the data as reported in Ref. 41, in terms of a mass-loss rate per unit area. In Ref. 41, the author states that the corrosion rates were calculated using the BET surface area. This approach is probably reasonable for the powdered materials, but is not appropriate for the coated beads. Both pyrocarbon and SiC have very small BET surface areas with very little surface-connected porosity. These materials are essentially impermeable at normal pressures and the exposed surface area is more accurately characterized by the geometric surface area. For assessing performance of a coating in a repository environment, a layer thinning rate (e.g., in the units μm/yr) is needed. The fractional mass-loss rate (FMLR) can be obtained from the Ref. 41 data by multiplying the data by the BET surface area. The layer thinning rate (LTR) is then obtained by dividing the FMLR by the product of material density (ρ) and geometric surface area per unit mass (A_{sm}):

$$LTR = \frac{FMLR}{\rho A_{sm}} \quad . \tag{6-1}$$

Unfortunately, the author of Ref. 41 did not provide the material densities or the geometrical parameters to determine A_{sm}, other than stating that the coated-particle diameters were approximately 500 μm. If it is assumed that a 500-μm particle was coated with 30 μm of pyrocarbon of density 1.9 g/cm^3, the quantity A_{sm} is 190 cm^2/g, which is about the same as the BET surface areas reported in Ref. 41 for pyrocarbons (see Table 6-1). Hence, it is reasonable to estimate LTR by using the data as reported in Ref. 41 and dividing by material density. Using this approach, Fig. 6-2 shows the time required to completely corrode a 40-μm thickness of the three materials described in Table 6-1. These results show that extremely long time periods would be required to corrode these materials, unless fuel and graphite temperatures remained elevated for very long time periods, which should not be the case (see Fig. 5-5). This calculation also presupposes the materials are continuously exposed to water over geologic time scales, which should also not be the case.

The Ref. 41 data also show an apparent correlation with heat-treatment temperature. Pyrocarbons, which are typically deposited at 1000 to 1300°C, exhibited corrosion rates similar to that of GC-1000. Graphite, which is typically heat treated at temperatures greater than 2000°C as part of the manufacturing process, exhibited a corrosion rate similar to that of GC-2600. This result is not unexpected, since greater outgassing occurs at higher heat treatment temperatures and the corrosion rates were inferred from CO and CO_2 measurements. Also, the corrosion rate could depend on crystallite size and structure, which can be affected by heat treatment. As part of the fuel manufacturing process, GT-MHR fuel compacts are heat treated to approximately 1800°C. Hence, the Ref. 41 graphite corrosion rates may be representative of the corrosion behavior of both graphite and pyrocarbon in GT-MHR spent fuel elements.

The ORNL and PNL data are consistent with data obtained by the Germans (Ref. 42). The Germans tested the leach resistance of fuel spheres and TRISO-coated particles at temperatures up to 150ºC and pressures up to 300 atmospheres in brine for exposure times exceeding four years. Quantitative corrosion rates were not reported, but the TRISO coatings showed no indication of corrosion.

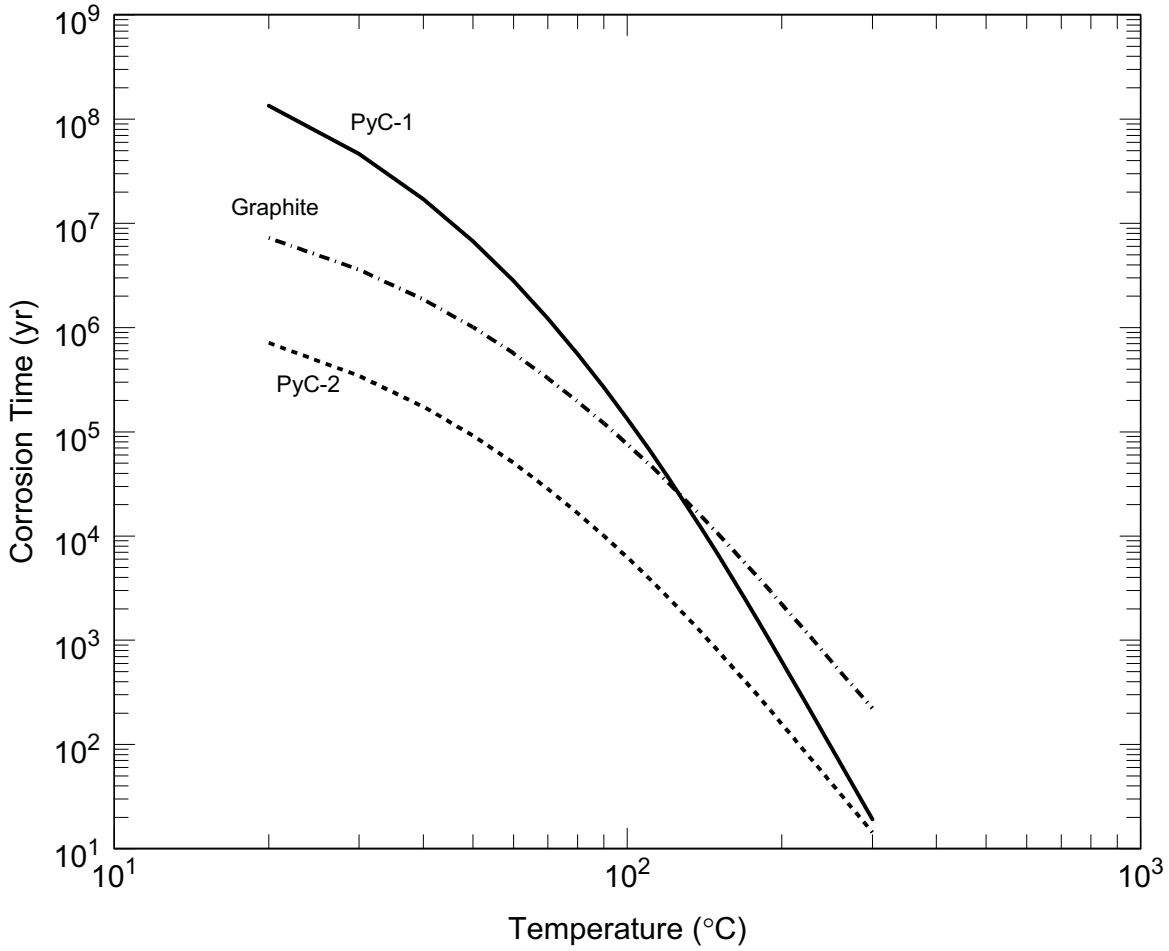

Figure 6-2. Time Required to Corrode a 40-μm Thickness

6.2 Long-Term Performance of Coated-Particle Fuel

For the PC-MHR, it was shown the coated particles should remain intact over geologic time scales (Ref. 6). The calculations were based on a simplified stress model, accounting for internal pressure buildup from radiogenic helium, a statistical distribution in the strength of the SiC layer (which functions as a pressure vessel), and continuous corrosion of the TRISO coating by groundwater. The corrosion resistance of the outer pyrocarbon layer was conservatively neglected, and the SiC corrosion rate of 1.7×10^{-5} μm/yr was based on data for natural glasses (Ref. 43). Using the corrosion-rate parameters in Table 6-1 for graphite and converting to a linear corrosion rate, graphite is predicted to corrode at this same rate at a temperature of 37°C.

A closed-form expression for the failure probability is derived by comparing a simplified, point calculation of the SiC hoop stress with a Weibull distribution of the SiC strength (Ref. 44). The failure probability P_F is calculated according to

$$P_F = 1 - \exp\left[-\ln 2 \left(\sigma / \sigma_f \right)^M \right] \quad , \tag{6-2}$$

where $\sigma \equiv$ maximum circumferential hoop stress in the SiC layer, $\sigma_f \equiv$ median fracture strength of the SiC, and $M \equiv$ Weibull parameter. The values σ_f = 388 MPa and M = 7.9 are used, based on an empirical fit to data obtained from heating TRISO-coated particles that were developed for the production of tritium (Ref. 6). The quantity σ is calculated according to

$$\sigma = \frac{P R_{SiC}}{2 t_{SiC}} \quad , \tag{6-3}$$

where $P \equiv$ internal pressure acting on the inner surface of the SiC layer (accounting for stable fission gases and helium buildup from alpha decay), $R_{SiC} \equiv$ radius to the middle of the SiC layer, and $t_{SiC} \equiv$ SiC thickness. The ideal gas law is used to calculate P, accounting for the void space within the kernel and buffer. For wet conditions, the quantities R_{SiC} and t_{SiC} are decreased with time as the result of SiC corrosion. This model is incorporated into the NEFREL module of the REPPER code (Ref. 8).

Figure 6-3 shows the buildup of gas inventory from radiogenic helium within GT-MHR fissile and fertile particles.* The buildup in gas inventory is more than compensated by the lower temperatures in the repository, such that the internal pressure in the particle is less than that during high-temperature irradiation. The corresponding failure probabilities are several orders of magnitude below the allowable core-average failure fraction of 5×10^{-5} during normal operation (see Table 3-2). This high-level of performance would not be compromised if the repository were to remain relatively dry with only sporadic contact of the waste packages by water.

* Because of limitations associated with the Gargoyle code (Ref. 24), the decay of actinides over very long time periods and the resulting buildup of helium has not been performed for GT-MHR fissile and fertile fuel. The data for the curves shown on Fig. 6-3 were obtained by scaling results for PC-MHR fuel according to discharged actinide inventories in the particles. The scaling factors were 0.36 for fissile fuel and 0.45 for fertile fuel.

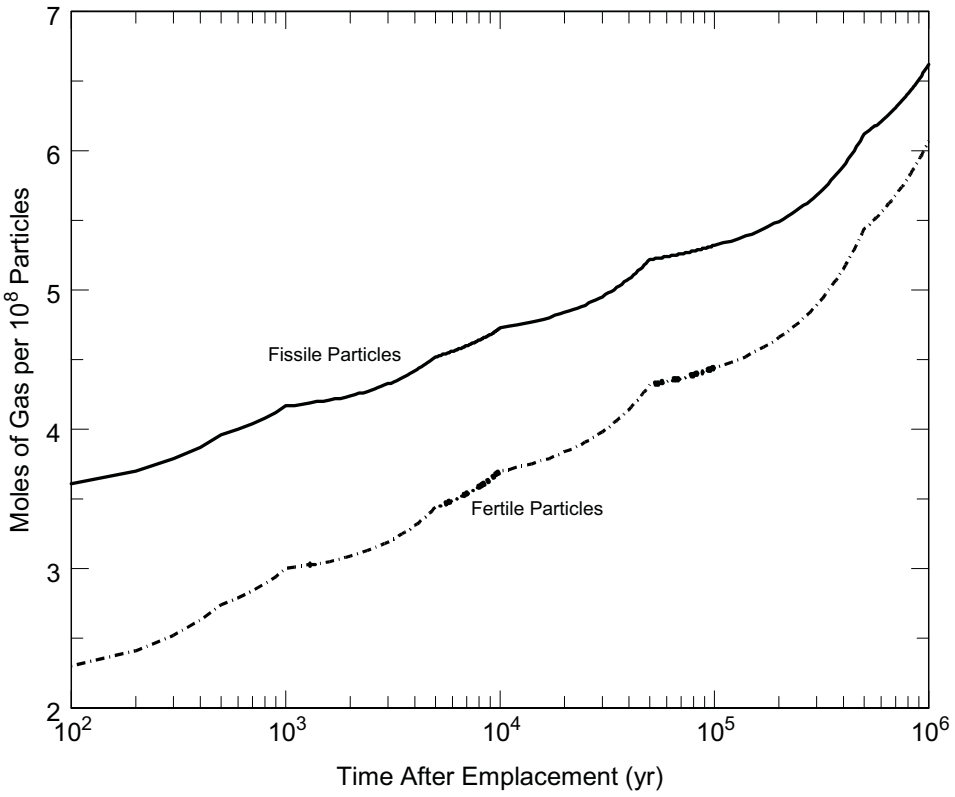

Figure 6-3. Buildup of Gas Inventory from Radiogenic Helium

Although DOE has dismissed the assessment of repository flooding as a disruptive scenario for disposal of CSNF and defense high-level waste (see Section 2.8), it is of some interest to examine this scenario for disposal of GT-MHR spent fuel. NEFREL models for particle performance were developed to assess this scenario based on the following assumptions:

- The repository becomes flooded with groundwater at 500 years after emplacement and remains flooded from that point on.

- Over the next 100 years, the canisters are assumed to fail and cool to a groundwater temperature of 30ºC.

Two cases were considered:

1. No credit is taken for the corrosion resistance of the OPyC. The SiC is assumed to start corroding at 600 years at a rate of 10^{-5} μm/yr. This corrosion rate is equivalent to the linear corrosion rate for graphite at 30ºC. As shown on Fig. 6-1, the corrosion rate for graphite falls between the rates for PyC-1 and PyC-2 at lower temperatures.

2. The corrosion rate is assumed to be an order of magnitude higher at 10^{-4} μm/yr, which corresponds to the rate for PyC-2 at 30ºC. However, because this corrosion rate is likely to be near the upper limit for GT-MHR particle coatings, credit is taken for the corrosion resistance of the OPyC. For a 40-μm thickness, the OPyC would protect the SiC for $40/10^{-4}$ = 400,000 years. At this point, the SiC is assumed to start corroding.

Figure 6-4 shows the reduction in SiC thickness for the two cases. For Case 1, nearly 100,000 years of continuous exposure to water is required to remove 1 μm of the SiC layer. After 1 million years, approximately 10 μm of SiC have been removed. However, even at this reduced thickness, the hoop stress in the SiC layer is still much lower than the fracture strength, and the predicted failure probabilities are negligibly small. For Case 2, the SiC layer is completely removed after 750,000 years (400,000 years to remove the 40-μm OPyC plus 350,000 years to remove the 35-μm SiC). As shown on Fig. 6-5, there is not much difference in the performance behaviors of fissile and fertile fuel for this case. The failure fraction exceeds 10^{-5} after about 675,000 years and eventually reaches unity at about 730,000 years. Because of radioactive decay over this very long time period, the radiotoxicity of the spent fuel has been reduced by 3 to 4 orders of magnitude (see Fig 3-5) before any significant levels of radionuclide release could occur. For comparison, approximately 50% of the CSNF waste packages are predicted to have failed by 100,000 years for non-flooded repository conditions (see Fig. 6-6).

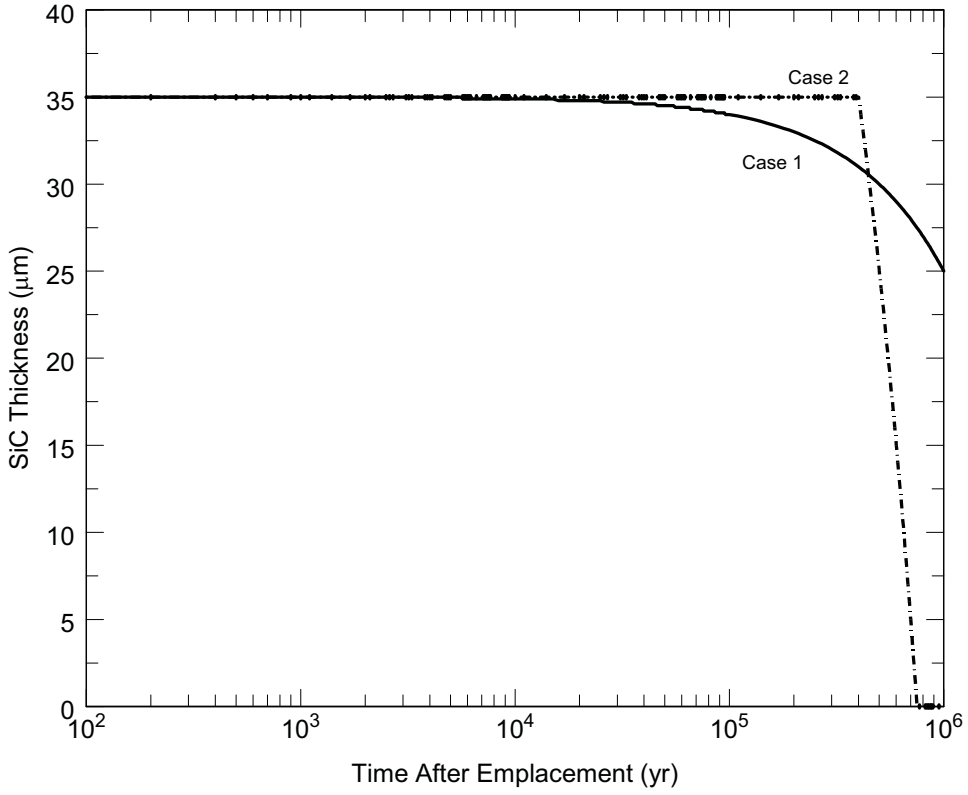

Figure 6-4. Corrosion of the SiC Layer

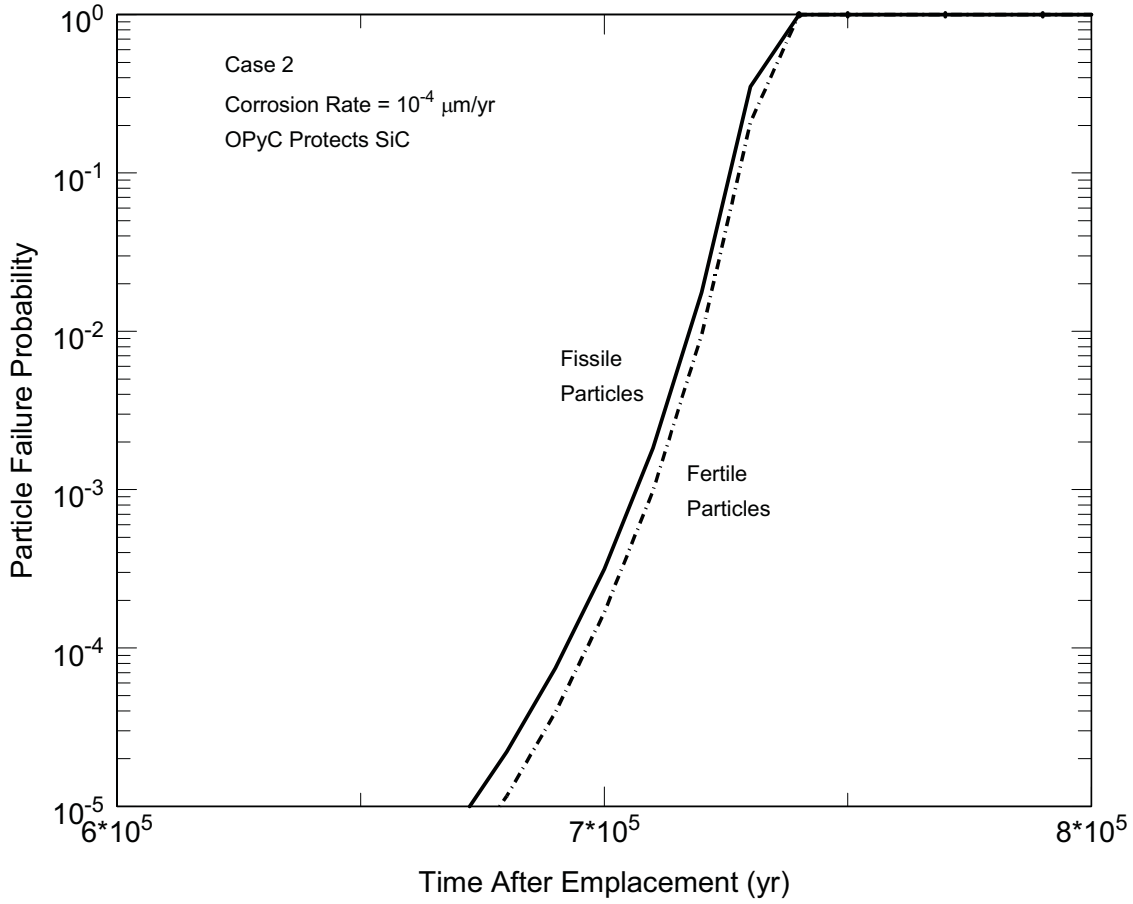

Figure 6-5. Predicted Failure Fractions for Case 2

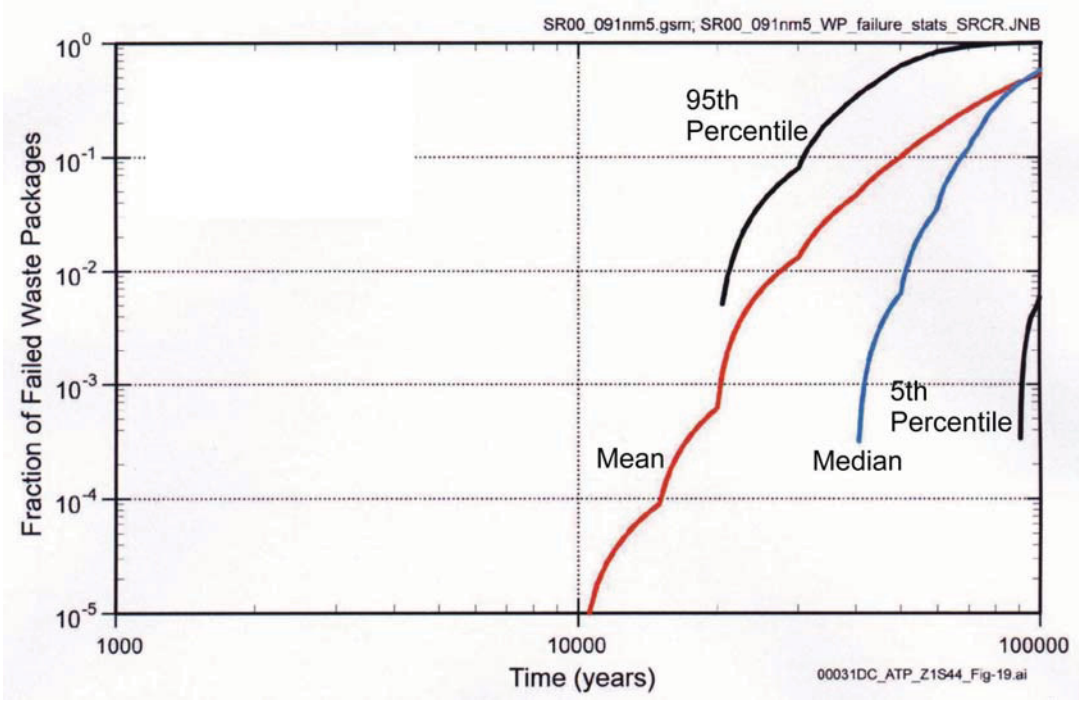

Figure 6-6. Performance of CSNF Waste Packages for Non-Flooded Repository Conditions (reproduced from Ref. 1)

6.3 Long-Term Performance of Graphite Fuel Blocks

Even though the graphite itself could be classified as Class C, low-level waste, it would likely represent the largest potential source term for assessing repository performance of GT-MHR spent fuel, primarily because of its relatively high C-14 content (see Section 3.5). Because the bulk of the C-14 inventory is produced by the $^{14}N(n,p)^{14}C$ reaction, C-14 (and other activation products) can be released at a rate higher than the corrosion rate of the base material. Several tests have been performed to estimate the radionuclide leach rates from nuclear-grade graphites. Key results from these tests are summarized in the following paragraphs.

The Germans have studied the leach behavior of pebble fuel elements previously irradiated in the AVR (Ref. 45). The leach tests were conducted in a synthetic brine solution for up to 464 days. Both the brine and gases vented from the top of the experimental rig were analyzed on a routine basis (approximately once per month). The following results were obtained:

- No C-14 was detected in the vented gases.
- C-14 released to the brine was chemically separated into organic and inorganic forms. Release of organic C-14 was measured throughout the test, with the release rate diminishing with time. Inorganic C-14 was released only near the beginning of the test.

- The cumulative fractional release of C-14 inventory was about 10^{-4}. The fractional C-14 release rates near the end of the test were approximately 10^{-5}/yr.

The British have performed graphite leaching tests as part of decommissioning studies for the Magnox and Advanced Gas Reactors (Ref. 26). The tests were performed on small graphite samples that were irradiated for 13 years in a CO_2-cooled Magnox reactor. The exposure conditions are not clearly defined in Ref. 26, and it is not clear if the samples had been sealed from oxidation by the CO_2 coolant. However, prior to leaching, the samples were machined to remove the surface layer. Pre-test measurements showed significant quantities of H-3, C-14, Fe-55, Co-60, Ba-133, Cs-134, Eu-154, and Eu-155 in the graphite. Leach tests were performed using both groundwater and seawater simulants. The groundwater tests were performed at 25ºC and a pressure of 1 bar (0.987 atm) for an exposure time of 150 days. The following nuclides were detected in the leachate samples: H-3, C-14, Co-60, Ba-133, and Cs-134. Activities of other nuclides were too low to be detected. The cumulative release of all nuclides showed an initial sharp increase over the first 10 to 20 days of leaching, which was attributed to rapid release of nuclides on the graphite surface. Except for Co-60, the cumulative release of all nuclides leveled off after 40 to 60 days of exposure. The cumulative release of Co-60 continued to show a steady increase at 150 days of exposure. Results for groundwater leaching are given in Table 6-2.

Table 6-2. Radionuclide Release from Leaching Magnox Reactor Graphite

Nuclide	Cumulative Release Fraction at 100 Days	Fractional Release Rate at 100 Days (yr^{-1})
H-3	0.0048	1.5×10^{-3}
C-14	0.0014	3.3×10^{-4}
Co-60	0.013	0.039
Ba-133	0.26	0.44
Cs-134	0.18	0.044

The leach behavior of graphite irradiated in a French Magnox reactor is described in Ref. 46. The leach tests were performed in deionized water for approximately 90 days and the results were compared with previous tests for graphite irradiated in the Hanford defense-program reactors. For Hanford graphite, the fractional release rate of C-14 was about 7×10^{-4}/yr, or about a factor of two higher than the rate measured by the British (Ref. 26). For the French graphite, the fractional C-14 release rates were much higher, ranging from 4×10^{-3}/yr to 0.03/yr. The higher leach rates observed for the French graphite may be caused by oxidation of the graphite by the CO_2 coolant during irradiation.

Because the GT-MHR uses high-purity helium coolant, very little oxidation occurs during irradiation, and the lower leach rates measured for British and Hanford graphite are probably more applicable. Nonetheless, if the canisters fail and the graphite is exposed continuously to groundwater, the C-14 inventory could be released over a time period that is relatively short compared with geologic time scales. For example, using the rate given in Table 6-2, the entire

C-14 inventory could be released in about 3,000 years.* The radiological consequences of C-14 release have been assessed for disposal of PC-MHR spent fuel (Ref. 12), and these results are directly applicable to disposal of GT-MHR spent fuel, since the differences in C-14 inventories are small. Release and transport calculations were performed using the REPPER code, using a model of the type shown in Fig. 6-7. Some of the calculations were performed using very conservative assumptions, including a flooded repository and neglecting retardation of C-14 during transport. Even with these assumptions, the predicted dose rates to an individual were very low and well below the regulatory criteria. The British also performed repository-performance assessments for their graphite and reached similar conclusions (Ref. 26). For example, for disposal at an inland site, the British calculated a maximum dose rate of 0.0057 mrem/yr. These results are not surprising and are consistent with the predicted radiological consequences of C-14 release from CSNF. Figure 6-8, which is reproduced from Ref. 1, shows the predicted critical organ mean dose rate for I-129 (thyroid), Tc-99 (gastrointestinal tract), and C-14 (fat). Based on these results, it is clear that the radiological consequences associated with C-14 release have essentially no impact on overall assessments of repository performance.

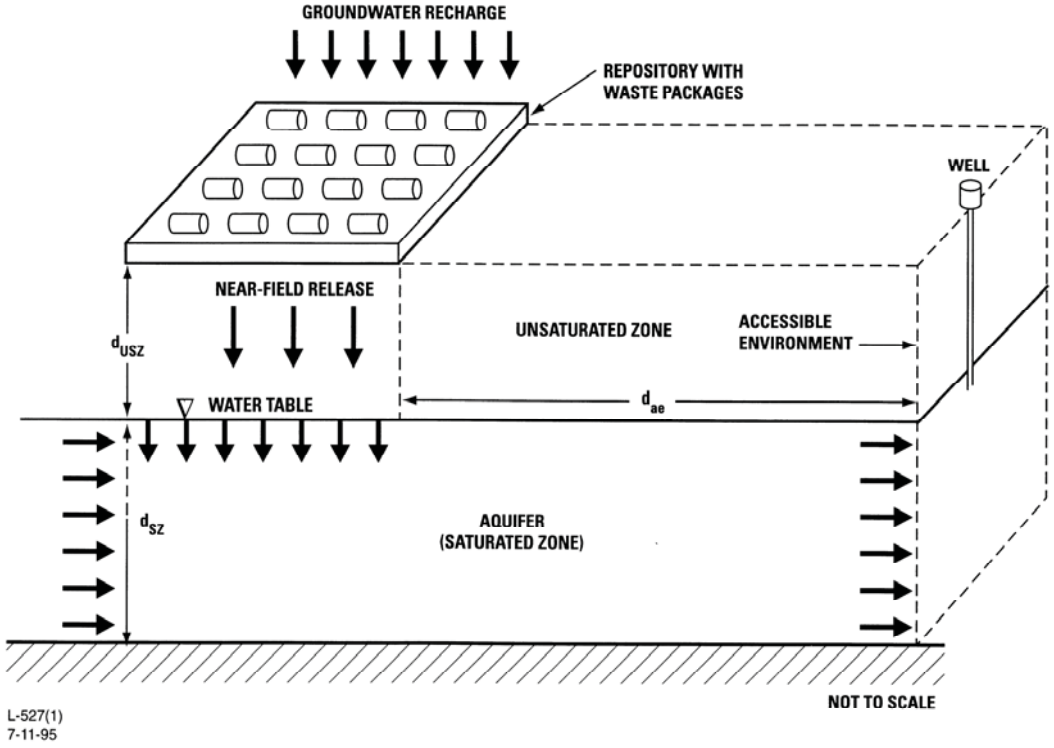

Figure 6-7. Repository Configuration and Groundwater Transport Pathways Modeled in REPPER

* The author speculates that not all of the C-14 inventory would leach at a relatively fast rate. The C-14 inventory produced by neutron activation of C-13 would be part of the graphite base material and would likely leach at a much slower rate.

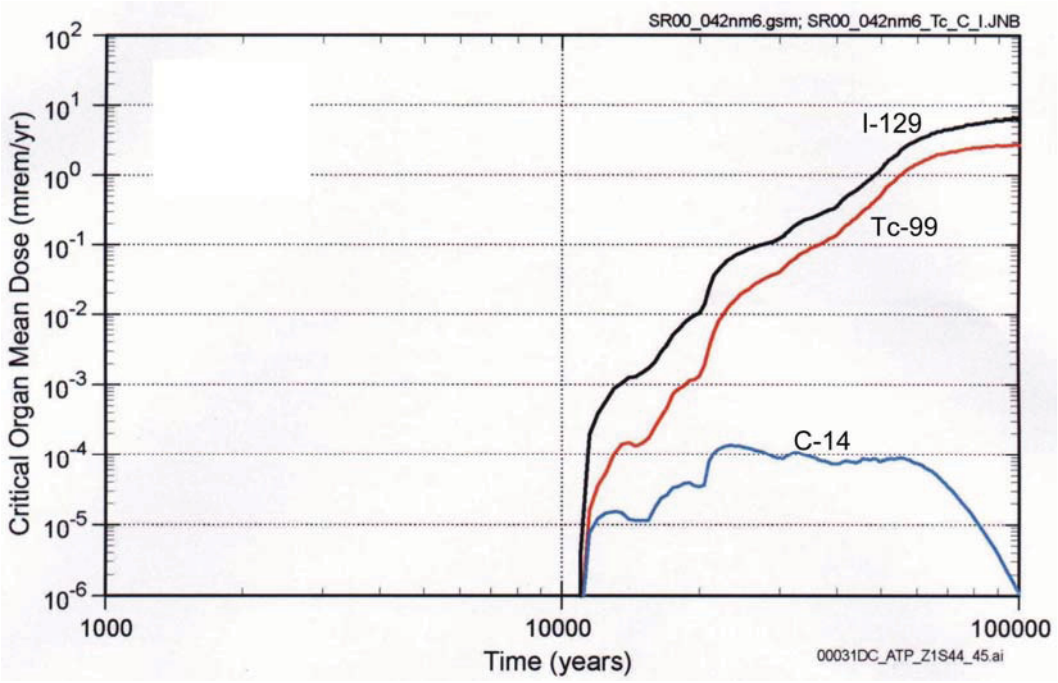

Figure 6-8. Mean Critical Organ Dose Rates from Disposal of CSNF at Yucca Mountain (reproduced from Ref. 1)

7. Conclusions and Recommendations

Based on this assessment, there are no technical issues or regulatory requirements that would preclude whole-element disposal of GT-MHR spent fuel in a geologic repository. In fact, GT-MHR spent fuel should be a nearly ideal waste form for permanent geologic disposal Specific features of GT-MHR spent fuel that support this conclusion include:

- The TRISO coatings provide a barrier that is highly resistant to groundwater attack. Based on the available data, the fuel particles should remain intact over geologic time scales, even if the repository were to become permanently flooded with groundwater.

- The nuclear-grade graphite blocks provide a structural container for the spent fuel that is also highly resistant to groundwater attack. Because of their very low impurity content, the graphite blocks with the fuel compacts removed would be classified as Class C, low-level waste. Leaching of the activation-product inventory from the graphite, including C-14, poses practically zero near-term or long-term radiological risk.

- Because of the GT-MHR's higher-burnup fuel cycle and high thermal efficiency, GT-MHR spent fuel contains significantly lower inventories of fission products and transuranic actinides than commercial LWR spent fuel, on a per unit electrical energy basis. These lower inventories translate to a lower IHI and a lower decay-heat load for GT-MHR spent fuel.

- GT-MHR spent fuel is highly resistant to proliferation and its characteristics are less favorable for recycle than CSNF. A GT-MHR MPC would contain only 2.7 kg of plutonium, which is a factor of 30 less than that for a typical PWR canister. Also, the plutonium isotopics in GT-MHR spent fuel are more degraded than those in CSNF.

- Because of the low power density and low fissile material content of GT-MHR spent fuel, its MPC design is relatively straightforward and inexpensive. Also, because of the robustness and high corrosion resistance of the spent fuel itself, there is no need to rely upon the GT-MHR MPC for long-term radionuclide containment. In contrast, the CSNF waste packages require the addition of neutron poison for criticality control and must include jackets of Alloy 22 and titanium drip shields to provide defense in depth from groundwater corrosion.

- For the reasons discussed above, the repository-loading strategy for GT-MHR MPCs is also relatively simple and straightforward. Up to 77 GT-MHR MPCs can be loaded per repository acre, and the MPCs can be surrounded with a low-permeability, highly-retarding overpack without compromising thermal design limits. In contrast, only 6 to 7 typical PWR waste packages could be loaded per repository acre, and even this sparse loading requires that the drift tunnels be actively cooled for 50 years after emplacement. Because of this active-cooling requirement, CSNF waste packages cannot be surrounded by an overpack, and an expensive titanium shield is required to provide additional resistance to corrosion and protection from rock fall.

This assessment of GT-MHR spent fuel disposal is highly encouraging and confirms previous results obtained for PC-MHR spent fuel. However, much additional work is needed to reduce uncertainties and increase confidence in predictions of long-term performance of GT-MHR spent fuel in a repository environment. To that end, a Confirmatory Test and Analysis Plan was developed for disposal of PC-MHR spent fuel (Ref. 7), and many of the research needs and test programs identified in that report are directly applicable to GT-MHR spent fuel. The corresponding plan for GT-MHR spent fuel should emphasize the following areas:

- Obtaining data for the corrosion rates of SiC, pyrocarbon, and nuclear-grade H-451 graphite. Ideally, the corrosion rates should be measured for irradiated materials.

- Obtaining data for the leach rates of radionuclides, especially C-14, from irradiated H-451 graphite.

- Obtaining data for the performance of candidate overpack/backfill materials. Although the long-term radiological consequences from geologic disposal of GT-MHR spent fuel should be practically zero without taking any credit for near-field retardation by overpack/backfill, the inclusion of this material will provide defense in depth and will address the multiple barrier approach specified in the proposed NRC 10CFR63 regulations. The material should effectively mitigate the transport of water to the waste package and effectively retard the transport of radionuclides from failed waste packages. Novel materials that can retard C-14 transport for the expected repository conditions, either through sorption or ion exchange, could provide some performance benefits and should be investigated.

In addition to these research and development needs, the MPC design should be developed further, with emphasis on reducing unit costs. Also, as recommended in Section 4.2, detailed criticality calculations need to be performed for an MPC loaded with GT-MHR spent fuel, in order to confirm the expected large design margins in k_{eff}.

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