

Repository Benefits of AFCI Options

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Executive Summary

Introduction

The results described in this report summarize the evaluations of potential benefits to a geologic repository for a variety of AFCI options that were studied during FY04. Many of the options were examined in response to a request by Burton Richter, chair of the Advanced Nuclear Transformation Technology (ANTT) Subcommittee of NERAC, to perform an initial evaluation of the potential benefit to a geologic repository from processing commercial spent nuclear fuel to separate certain chemical elements and to recycle some of these elements in thermal spectrum reactors such as light water reactors. The measure of repository benefit has been defined as the allowable increase in repository drift loading consistent with satisfying all repository thermal design limits, since loading of a geologic repository at Yucca Mountain is currently limited by temperature constraints. Such an increase in drift loading can be used to either reduce the size of a repository of given capacity, or to increase the capacity of a repository of a given size. Any changes in estimated peak dose caused by the resulting alteration in the radionuclide inventory of the repository have not been evaluated, but are the subject of a separate ongoing study.

Previous work has shown that there is a large potential for increasing the drift loading in a repository by removing and recycling plutonium and americium from spent LWR fuel, while also removing the cesium and strontium for separate storage.[ES1] The loading of a repository at Yucca Mountain is currently constrained by the temperature limit midway between adjacent drifts, set at the local boiling point of water of 96 °C, to allow water to flow down unimpeded through the repository at all times. The source of the heat responsible for approaching this temperature limit is the decay of plutonium and americium over the extended time period from the time of repository closure to 1500 years after waste placement, mainly from Am-241, Pu-238, Pu-240, and Pu-239, in order of importance. Removal of the plutonium and americium from commercial spent fuel allows the drift loading to be increased, by about a factor of 5-6 for a separation efficiency of 99.9%.

It might be expected that removal of 99.9% of the plutonium and americium would have resulted in a larger gain in drift loading, but using the current design and operating conditions for the repository, the gain is restricted due to the remaining materials in the emplaced waste generating sufficient decay heat in the near term to cause a different temperature limit to be reached as drift loading is increased.[ES1] Shortly after closure of the repository with the drift loading increased by about a factor of 5-6, the temperature of the drift wall approaches the limit of 200 °C established to prevent crystalline alteration of the rock. Since the plutonium and americium have already been removed, the source of the heat causing this temperature limit to be approached is barium and yttrium, decay products of cesium and strontium. Barium and yttrium each have a very short half-life, so removal of cesium and strontium is sufficient to address this issue. In principle, the subsequent removal of cesium and strontium, in addition to the plutonium and americium, allows for much larger gains in drift loading for the current repository design and

operating conditions, in excess of a factor of 40-50 with recovery efficiencies of 99.9% for these elements, while staying within thermal limits for waste packages, as well as the repository environment. However, previous work has also demonstrated that consideration of realistic processing and recycling schemes may reduce the magnitude of the benefit, depending on; 1) how completely the recovered plutonium and americium are separated and transmuted, 2) whether the elements are continuously separated and recycled, or are only recycled a finite number of times, and 3) on the composition and eventual disposition of all of the waste streams. The impact of these effects is quantitatively evaluated in this study.

Analysis Approach and Assumptions

Given this background, the focus of the repository benefit evaluations during FY04 was to examine specific recycling schemes using existing light-water reactors, and to quantify the potential repository benefit, using the following general guidelines and assumptions:

- Spent PWR fuel was used as the example of commercial spent nuclear fuel, as it represents the majority of material destined for a repository at Yucca Mountain, and is also packaged to the highest linear heat load in a repository drift.
- Plutonium and americium are separated from the spent PWR fuel with an efficiency of 99.9% to address the repository heat load issue. Neptunium is also separated from the spent PWR fuel for non-proliferation and radionuclide inventory concerns. Plutonium, americium, and neptunium are then recycled for further irradiation in an LWR.
- The spent fuel is assumed to be processed at 5 years after discharge to minimize the buildup of americium during storage and to maximize the potential benefit of recycling in a thermal spectrum to reduce the long-term heat load. The effect of delaying processing and recycling of spent fuel, processing at 20 years after discharge, has also been considered for a single recycle in each strategy to determine the sensitivity of the calculated repository benefit to the age of the spent fuel.
- The number of recycles of plutonium, americium, and neptunium is a parameter in the study. After the last recycling, all materials are sent to the repository, including both processing waste and the spent fuel assemblies from the last irradiation cycle.
- Curium is not separated from the spent PWR fuel, but is sent directly to the repository in the process waste stream.
- Cesium and strontium are separated from the processing waste, and are stored separately, either in dry storage external to the repository or in a separate area of the repository.
- The neutronic analyses of each assembly type, whether enriched uranium fuel or fuel containing the recycled elements, was performed using lattice calculations. This is roughly equivalent to treating the assemblies as if they were in reactor cores of identical assemblies, i.e., homogeneous reactor cores. In some cases, this leads to difficulties with reactor safety coefficients and other issues that may result in these cases being impractical, requiring the use of heterogeneous reactor cores. However, based on past results, it is expected that the results of the lattice calculations for spent fuel compositions would be very close to the results obtained for heterogeneous reactor cores.
- Assemblies fabricated using recycled material are considered as substitutes for additional standard PWR assemblies, and produce the same integrated energy per assembly. Groups of assemblies are formed consisting of the assemblies containing recycled materials and the corresponding amount of processing waste from previous generations of assemblies, representing a certain amount of total integrated energy. Each group is then compared

with a PWR group that produced the same total integrated energy. As a result, all determinations of repository benefit are performed on an **equal integrated energy basis**.

- The repository operation and design is consistent with the high-temperature operating mode (HTOM) of the cold repository, which is the current reference for a repository at Yucca Mountain. The repository drifts are 81 m apart, about 1 km in length, and 5.5 m in diameter. In this study, the repository is assumed to be ventilated for 75 years after repository closure to further reduce the importance of shorter-lived isotopes. This is longer than the minimum specification of 50 years, but within the operational envelope being considered by the Yucca Mountain Project. The thermal analyses calculating the response of a repository at Yucca Mountain are performed using the detailed 3-D model representing the central area of the repository where the highest temperatures are expected.[ES1] The relevant temperature limits for the HTOM have been discussed in the Introduction.

Thermal Reactor Recycling Strategies

For the purposes of this study, three LWR recycling strategies have been examined, identified as MOX, CORAIL-PNA, and IMF:

- MOX – Mixed-oxide fuel. In this approach, the separated plutonium, americium, and neptunium are used to fabricate new fuel assemblies, in a fuel matrix of recovered uranium (enrichment slightly above natural uranium), all elements being present as oxides. To obtain sufficient fissile material for the first generation of MOX, provided mostly by plutonium, it is necessary to use materials recovered from a number of spent PWR fuel assemblies to fabricate one MOX assembly. Subsequent recycling with MOX uses the recovered plutonium, americium, and neptunium from the current MOX generation to fabricate the next generation of MOX assemblies. Again, to provide sufficient fissile material, a number of spent MOX assemblies from one generation need to be processed to fabricate one MOX assembly in the next generation. For each parametric case on the number of recycles, the last generation of MOX assemblies is directly disposed to the repository, along with the processing waste from all previous generations.
- CORAIL-PNA – This concept uses heterogeneous assemblies, where some of the fuel pins (about 1/3) are fabricated from the separated plutonium, americium, and neptunium, in a uranium matrix, and the remaining fuel pins (about 2/3) are fabricated from new enriched uranium, with all elements present as oxides. In the CORAIL-PNA case, the recovered plutonium, americium, and neptunium from one spent PWR assembly is used to make the 1/3 of the fuel pins for one assembly in the first CORAIL-PNA generation that contain these materials. After irradiation, the entire spent CORAIL-PNA assembly is processed to recover the plutonium, americium, and neptunium for 1/3 of the fuel pins in a single assembly of the next CORAIL-PNA generation. The remaining 2/3 of the fuel pins are again fabricated from new enriched uranium, with the enrichment increasing with each CORAIL-PNA generation. For each parametric case on the number of recycles, the last generation of CORAIL-PNA assemblies is directly disposed to the repository, along with the processing waste from all previous generations.
- IMF – Inert Matrix Fuel. This approach is similar to MOX, but the fuel matrix is an inert material, zirconia, instead of uranium oxide. The recovered plutonium, americium, and neptunium from several spent PWR assemblies are used to make a single assembly in the first generation of IMF so that sufficient fissile material is provided. Subsequent

generations of IMF also use several IMF assemblies of the previous generation to obtain sufficient fissile material. As with the other cases, the number of recycles is a parameter in the study, and after the irradiation of the last generation in each case, the spent IMF assemblies are directly disposed to the repository along with the processing waste from all previous generations.

Repository Drift Loading Increases for Processing Spent Fuel at 5 Years

For each of these cases, detailed fabrication, separation, and irradiation histories had been calculated in a parallel effort by J.A. Stillman [ES2] and T.K. Kim [ES3] for a range of the number of recycles, with the assumption that all spent fuel was processed at 5 years after discharge to minimize buildup of americium from plutonium decay. As described in the previous section, the resulting isotopic compositions of the spent fuel assemblies and the processing waste were collected into the appropriate groups to preserve total integrated energy produced and compared with the reference case of direct disposal of spent PWR fuel. It should be noted that there are many issues associated with some of these strategies, including fuels development, designs of practical reactor cores, and fuel processing methods. These remain as open issues for further study and evaluation.

The potential increases in repository drift loading are displayed in Figure 1, with general results for each case as follows (where it is also noted that all cases shown in the figure may not be realizable in practice):

- MOX – The use of MOX fuel allows a steady increase in repository drift loading with each recycle of plutonium, americium, and neptunium, reaching a factor of 1.5 after 5 recycles of MOX fuel. The factor for the increase in drift loading is a combination of a very large factor for the process waste (~40) and a very small factor for the direct disposal of the spent MOX assemblies from the last recycle after the specified number of recycles (typically <0.10). At this time, it is not clear that the use of MOX to 5 recycles could be realized due to several difficulties, especially with reactor safety coefficients, but the use of heterogeneous reactor cores with assemblies of all generations may alleviate this problem, and would need to be studied further.
- CORAIL-PNA – The use of CORAIL-PNA also allows a steady increase in drift loading with each recycle at a faster rate than for MOX, reaching a factor of 2.0 after 7 recycles of CORAIL-PNA fuel due to the favorable impact of using enriched uranium to provide fissile content rather than relying entirely on the recovered plutonium, americium, and neptunium. As with MOX, the factor for increasing the drift loading is a combination of a very large factor for the process waste (~40) and a very small factor for the direct disposal of the spent CORAIL-PNA assemblies from the last recycle after the specified number of recycles (<0.10). Unlike the MOX cases, though, the use of enriched uranium to supplement the fissile content in the CORAIL-PNA concept appears to allow recycling to reach an equilibrium state, where the charge and discharge amounts of plutonium, americium, and neptunium are equal and only process losses would go to the repository.
- IMF – The use of inert matrix fuel provides a factor of 1.8 for the increase in drift loading after the first recycling, and a factor of 2.1 after the second recycling. Further recycle of IMF is hindered by the rapid depletion of fissile material with each subsequent irradiation (especially for Pu-239), and makes it impossible to perform a third recycle to the same integrated energy for the assembly. Additional recycling using IMF fuel may be possible using heterogeneous assembly designs and blending with other fissile material, but this

variation has not yet been evaluated. As with the other cases, the factor for increasing the drift loading is a combination of a very large factor for the process waste (~40) and a very small factor for the direct disposal of the spent IMF assemblies from the last recycle after the specified number of recycles (<0.10).

- In all cases - The recycling of neptunium appeared to be detrimental to increasing the drift loading in the repository, mainly through neutron capture in Np-237 to create additional heat-generating Pu-238. While not the dominant isotope, Pu-238 is second in importance to Am-241 in providing decay heat and limiting the drift loading of the repository. Some increase in benefit is expected if only plutonium and americium are recycled, but the amount has not been quantified at this time.

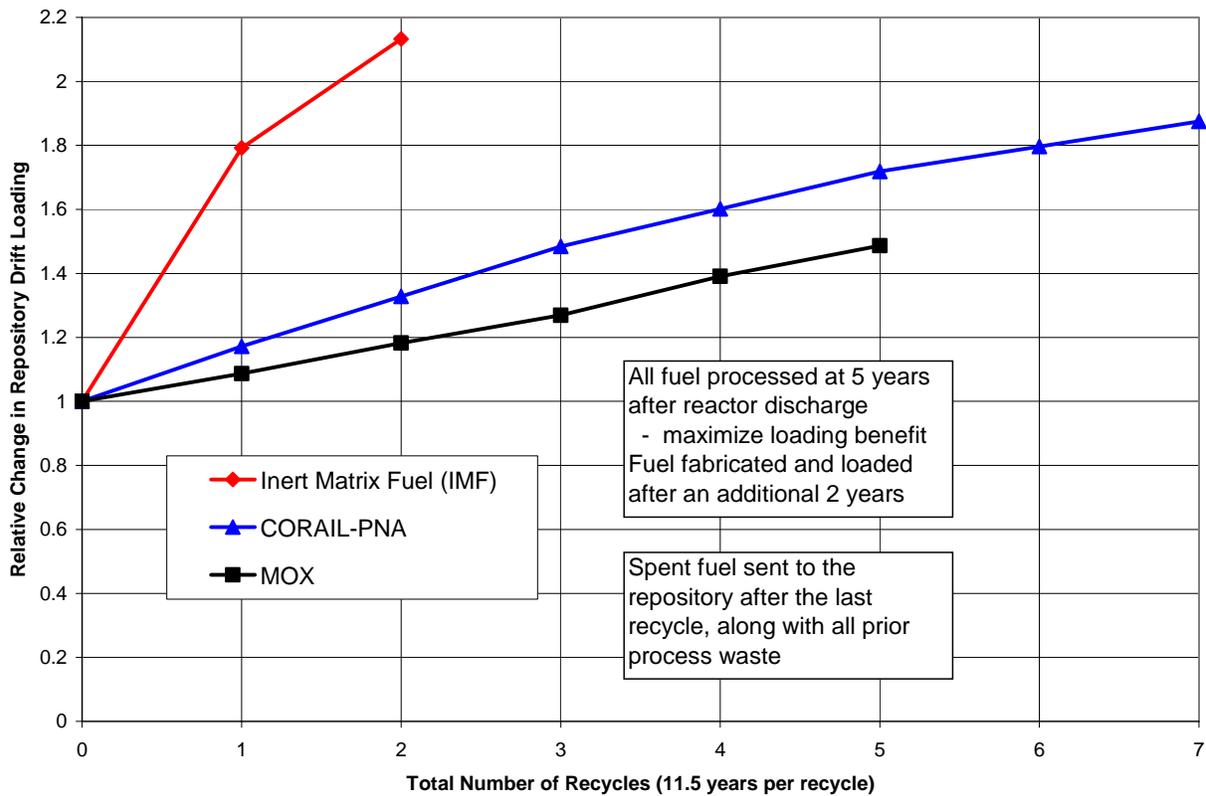


Figure 1. Relative change in repository drift loading as a function of the number of recycles for MOX, CORAIL-PNA, and IMF recycling options, fuel processed at 5 years after reactor discharge, with spent fuel sent to the repository after the last recycle.

Sensitivity of Repository Drift Loading Increase to the Age of the Spent Fuel

Since the repository drift loading for direct disposal of spent PWR fuel is controlled by the decay heat from americium (Am-241) arising from the decay of Pu-241 in the stored spent fuel after irradiation, the analyses reported in the previous section were performed for processing all spent fuel at 5 years after discharge to minimize the decay of Pu-241 and thus maximize the benefit to the repository. The benefit to the repository is also closely related to the total amount of Pu-241 (and Am-241) in the discharged fuel when the spent fuel is directly disposed in the repository. Higher enrichments are needed when older fuel is processed and recycled due to the decay of fissionable isotopes and the increase in isotopes that are more likely to capture neutrons in the

thermal spectrum. The higher enrichment increases both the Pu-239 and Pu-240 content in the recycled fuel, and the net production of Pu-241 is increased, potentially reducing the benefit to the repository from processing and recycling spent fuel. To quantify the sensitivity of the repository drift loading increase to the age of the spent fuel, analyses were also conducted for one recycle in each case using spent fuel that had been stored for 20 years. (This part of the study was performed subsequent to the initial report to the ANTT Subcommittee of NERAC, and was related to ANTT Subcommittee questions on the effects of extended cooling prior to processing. The effort in this area was covered by a separate AFCI work package, but is included in this report for completeness of description of the repository benefits evaluations.) The results are summarized in Figure 2.

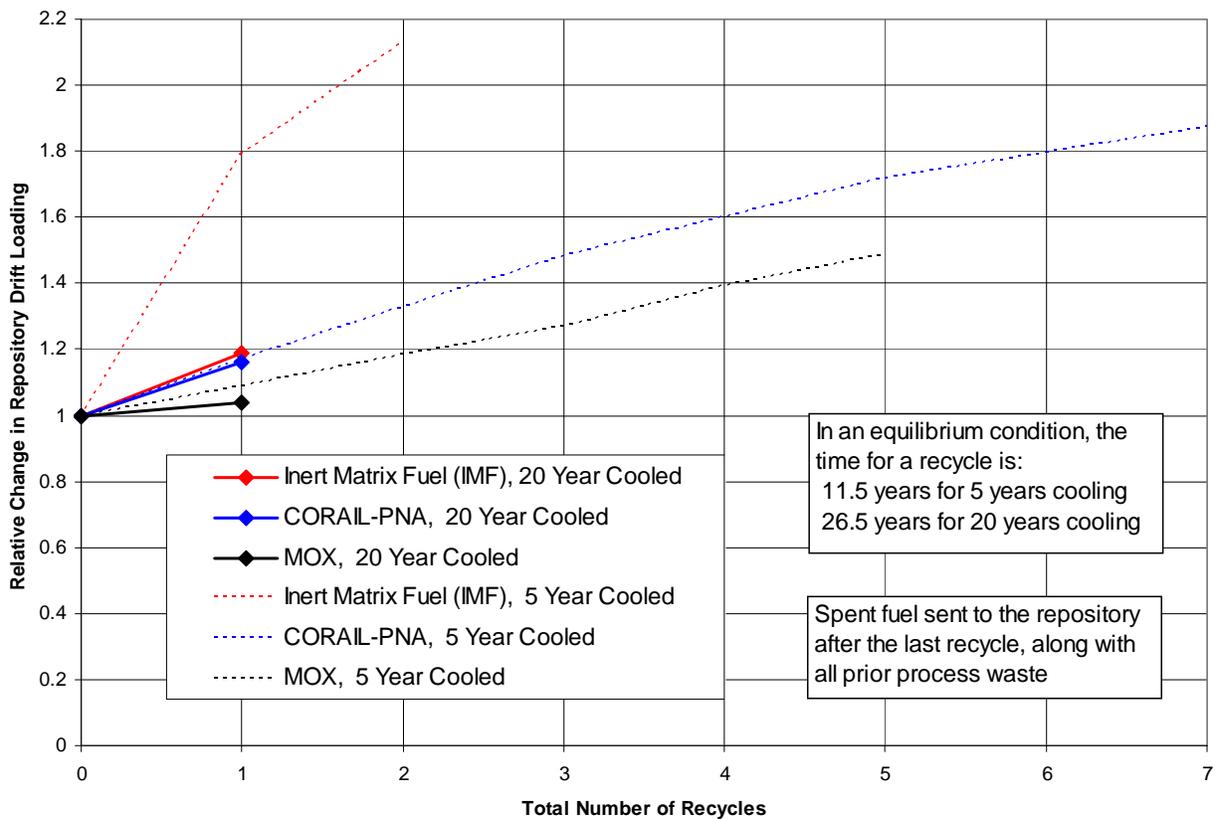


Figure 2. Relative change in repository drift loading as a function of the number of recycles for MOX, CORAIL-PNA, and IMF recycling options, fuel processed at 20 years after reactor discharge, with spent fuel sent to the repository after the last recycle.

As is shown in the figure, the effect of processing older fuel and recycling the recovered plutonium, americium, and neptunium in an LWR differs depending on the recycling strategy. The least impact is observed for CORAIL-PNA, where the use of enriched uranium reduces sensitivity to the changing isotopic composition of the recycled materials by allowing lower plutonium enrichment. The reduction in benefit as measured by the increase in repository drift loading is slightly less than 30%. The greatest impact is observed with IMF, since IMF depends entirely on the fissile content of the recovered materials, which is mostly plutonium, and more fissile is needed as the composition and isotopic distribution changes with age. In this case, the reduction in drift loading benefit is over 75%. The effect on MOX is between these two, with a

reduction in the allowable drift loading increase of about 50%, due to the use of a uranium matrix that provides additional fissile material during irradiation. However, the overall effect is that for a single recycle step using older fuel, the use of IMF is comparable to using CORAIL-PNA, with the benefit of using MOX becoming minimal.

Remaining Issues and Considerations

As additional comments, it is useful to note that processing and recycling spent PWR fuel could also be used to lower temperatures in a geologic repository, maintaining the reference drift loading as currently planned for a repository at Yucca Mountain. This would allow more stringent temperature limits to be met, if needed, and could be used to provide greater assurance about the performance of the repository. Also, the reduction in the radionuclide inventory and the resulting alteration in estimates of the peak dose rate associated with releases from the repository have not been specifically addressed in this study, since this is not an issue for a repository at Yucca Mountain at this time as dose rate from releases does not constrain repository loading. However, some of the elements that are responsible for producing the decay heat in a repository (Pu, Am, Cs, and Sr), plus Np, are the main contributors to the dose rate. In a related ongoing study, quantifying the effects of the processing and recycling strategies using detailed repository performance assessment is showing that substantial dose rate reductions may be possible.

Systematic PWR fuel processing and recycling as evaluated in the current study would appear to offer other potential benefits for nuclear power, such as being able to densely dispose of process waste in a repository while greatly reducing the number of irradiated fuel assemblies that would need to be stored at reactor sites. Such possibilities emphasize the need to consider all relevant advantages and disadvantages when evaluating the usefulness of recycling in thermal reactors.

Conclusions on the Impact of Limited Thermal Reactor Recycling

Results of this study have quantified the benefit to a repository as measured by increased drift loading of the repository; about a factor of 2 for the limited number of recycles considered. The results have also emphasized that approaches using a limited number of recycles will provide only a small fraction of the potential benefit (in excess of a factor of 40) that could be achieved with continuous recycling where the plutonium, americium, and neptunium remain in the fuel cycle, except for process losses. As discussed above, the general reasons for this have been determined as follows:

- In each case, the process waste is capable of being very densely loaded in the repository drift while still satisfying all thermal limits, about a factor of 40-50 greater than for spent PWR fuel. This is consistent with the previous results obtained for continuous recycling, since only process waste is sent to the repository in that case as well. (This can be increased to about a factor of 100 if curium were also separated from the process waste.)
- Direct disposal of the last assemblies in each recycling strategy requires most of the repository space, with allowable drift loading densities less than 5-10% that for spent PWR fuel. This is not unexpected, since all of the higher actinide elements from many assemblies have been concentrated in the assemblies containing recycled material.
- It is the combination of the increased drift loading density for the process waste and the reduced drift loading density of the last assemblies in each case that limits the overall benefit to the loading of the repository to a factor of about 2.

- Continuous recycling is essential for obtaining large increases in drift loading in the repository, and may be possible in a thermal spectrum using the CORAIL concept, or advanced MOX, IMF, or target strategies. The key is to prevent a large fraction (>99%) of the heat-producing transuranic inventory from ever being placed in the repository.

Given these conclusions, it would be useful to further quantify the impact of recycling scenarios using thermal neutron systems, where production of the higher actinide isotopes is enhanced, as compared with alternate scenarios involving irradiation in a fast neutron spectrum, where production of higher isotopes is limited.

However, the results of this study should not be interpreted to mean that processing spent PWR fuel and recycling in LWRs is of little or no benefit:

- By processing the spent fuel and sending only process waste to the repository, substantial increases in loading (or corresponding decreases in repository size) are possible as soon as the activities are begun.
- As long as the recycled plutonium, americium, and neptunium are kept in the fuel cycle, the large allowable increase in drift loading, ~ 40 or more, is realized and maintained. The need to keep the plutonium, americium, and neptunium in the fuel cycle should not necessarily be viewed as a disadvantage, since in any version of a uranium-fueled nuclear future where nuclear power generation is at least maintained, the eventual deployment of fast reactors appears to be inevitable and would provide the eventual destination for the remaining plutonium, americium, and neptunium. It should also be noted that in the absence of a nuclear future, processing of spent nuclear fuel from existing reactors is likely to be unnecessary, as sufficient repository capacity is likely to be available even within a repository at Yucca Mountain to store all of the spent fuel.
- The use of LWRs for recycling, where they are essentially plutonium burners, reduces the number of fast reactors needed to provide the continuing recycling of higher actinides for a given number of LWRs, as shown in previous work. [ES4]

In summary, it has been shown that processing spent PWR fuel and using limited recycling in LWRs is of modest benefit to a geologic repository where loading is determined by thermal constraints. In addition, consideration of older fuel virtually eliminates any advantage from novel fuel types such as inert matrix fuel, and that use of mixed-oxide fuel provides essentially the same benefit. However, it has also been shown that recycling in LWRs can be very beneficial as part of an overall strategy where nuclear power generation continues into the future.

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I. Background

In order to evaluate the impact of processing and recycling spent nuclear fuel on a repository at Yucca Mountain, it is essential to understand the reasons for the current design of the repository and the limiting conditions constraining the placement of waste or spent fuel. At this time, a repository at Yucca Mountain is being planned for the permanent disposal of spent nuclear fuel and high-level nuclear waste. Safe storage of hazardous radioactive materials must be guaranteed for long periods of time in order to limit any releases to the environment, where the peak dose rates associated with such releases are specified by regulation. In general, to adequately perform the mission of isolating hazardous radioactive materials from the environment, a geologic repository needs to satisfy a number of technical requirements concerning the state of the repository and the materials it contains so that projections about the overall performance of the repository would have a reasonable probability of being accurate. The technical requirements are usually related to the tested corrosion, degradation, and release characteristics of the repository materials for relevant ranges of environmental conditions, including temperature and water chemistry, and can take the form of specifications for peak allowable temperatures for the repository, specific materials to be used in the construction of disposal containers (packages), etc. In the case of a repository at Yucca Mountain, these considerations lead to the imposition of a number of temperature limits for various parts of the repository system, given that the waste disposal packages are constructed from a specific set of materials. These temperature limits result in specifications for the maximum decay heat of each waste package at the time of placement, and the maximum average linear heat rate for the array of waste packages in a repository drift (tunnel). [1]

The current reference operating mode of a repository at Yucca Mountain, the high-temperature operating mode (HTOM) of the “cold” repository, is calculated to have extended time periods where the temperature of the rock surrounding the waste storage drifts will be above the boiling

point of water. However, one temperature limit specifies that the rock temperature midway between adjacent drifts must always remain below 96 °C. This specification ensures that any water flowing downward through the mountain will be able to move through the repository at all times, preventing the retention of a large volume of water above the repository that could flood the repository once the temperatures have dropped. Another temperature limit is the specification that the rock temperature must remain below 200 °C at all times to prevent alteration of the crystalline structure of the rock. Both of these temperature limits are examples that are used to provide greater certainty about the conditions in the repository, increasing the reliability of the assessments of repository performance. Other temperature limits for a repository at Yucca Mountain apply to the emplaced materials and the waste packages to limit the degradation rate that leads to release of radioactive materials.

The temperature limits act as constraints on the design and operation of a geologic repository, and meeting these limits can be accomplished by a variety of methods, including controlling the amount of waste in any given area of the repository or actively cooling the repository for an extended period of time. The result for a repository at Yucca Mountain is that the emplacement drifts are 81 m apart at present, loaded to an average drift linear heat load of about 1.45 kW/m at the time of waste placement (assuming an average age of about 25 years for spent fuel, this corresponds to an average linear loading of spent fuel in the drift of about 1.1 MTIHM/m), and the repository is actively cooled by forced ventilation for a period of at least 50 years after the completion of the waste placement phase of operations.

I.A Separation of Chemical Elements and Potential Repository Benefit

In a previous study, the repository benefits of separating certain chemical elements from spent fuel and recycling them to alter the characteristics of the waste stream destined for a repository at Yucca Mountain were quantified.[2] The conclusions from that study are as follows:

1. The dominant contributors to the thermal load from the emplaced spent fuel or waste in a repository at Yucca Mountain that lead to reaching one or more of the temperature limits are plutonium and americium. Removal of these chemical elements, and recycling and transmuting them to reduce the inventory of these elements in the materials placed in the repository, is essential to increasing the repository drift loading. The benefit ranges from a factor of 5-6 in increasing the drift loading (or decreasing the repository size for a given capacity) with waste placed in the repository at 25 years after reactor discharge.
2. After the plutonium and americium have been removed, the next chemical elements that need to be considered are cesium and strontium, mainly for the heat produced by their short-lived decay products, barium and yttrium. Removing cesium and strontium, and sequestering them in a separate area of the repository or in another facility, would allow a substantial increase in the repository drift loading, up to a factor of 40-50 greater than the direct disposal case, for 99.9% removal of plutonium, americium, cesium, and strontium, and when the waste is placed in the repository 25 years after reactor discharge.
3. The next most important chemical element is curium. However, in considering a realistic recycling scenario for the plutonium and americium, assumed to be irradiation in a fast reactor, it was observed that the drift loading can be limited by the losses of these two elements from processing the spent fast reactor fuel. The 1% loss assumed in the processing of the fast reactor fuel reduced the potential increase in drift loading from a factor of 43 to 21, and dominated the decay heat generation. This emphasizes the need to

reduce recycling losses of plutonium and americium below the 1% assumed in this study before separation of curium will be effective.

4. The issue of peak dose rate can be addressed by removing and recycling the plutonium, americium, and neptunium, as these chemical elements are the dominant contributors to the peak dose rate. It should be noted that the estimated peak dose rate of 100 to 400 mrem/year is occurring at times well beyond the current 10,000 year regulatory period, about 250,000 years. Current estimates of repository performance indicate that the peak dose rate within the regulatory period is several orders of magnitude below the 15 mrem/year limit. Whether separations should be done to alter the estimated performance past the regulatory period appears to be an open question, although the potential for increased repository capacity would imply a need to evaluate the impact on the estimated peak dose rate to ensure that regulatory requirements are still satisfied.

It has been shown that removal of plutonium and americium has the potential for reducing the size of a repository at Yucca Mountain by a factor of 5-6. Combining this with removal of cesium and strontium allows for much greater reductions in size, upwards of a factor of 40, although the use of realistic recycling options for the plutonium and americium emphasizes the need to effectively transmute these elements and to have very low losses for processing the recycled fuel, regardless of the manner in which the recycled material is treated (thermal reactor, fast reactor, etc.) To take advantage of such a potential increase in drift loading requires the availability of waste forms that could be densely loaded with the remaining waste materials.

Examination of an alternative low temperature operating mode (LTOM) for a repository at Yucca Mountain shows that the same processing and recycling strategies would be effective for that case as well, yielding similar benefits. In summary, the study quantified benefits to a geologic repository that arise from certain spent fuel processing strategies, as would be examined as part of the AFCI program, with these benefits appearing to be relatively independent of the specific repository design decisions being considered for a repository at Yucca Mountain.[2]

I.B The AFCI Program and Thermal Reactor Recycling

In the Advanced Fuel Cycle Initiative (AFCI) program sponsored by the U.S. Department of Energy, numerous reactor, processing, and recycling strategies are being examined to determine the impact on issues important to the viability of nuclear electricity generation, including the disposal of spent nuclear fuel and nuclear waste. As part of this program, studies are being performed to determine the benefit of processing spent nuclear fuel, which separates certain chemical elements followed by transmutation of these elements, to geologic repositories by altering the decay heat of the emplaced waste and lowering the radionuclide inventory.

In this report, three strategies have been examined to quantify the impact that recycling in thermal neutron spectrum (light-water moderated) systems would provide to the repository; MOX (mixed-oxide fuel), CORAIL-PNA (heterogeneous mixed-oxide/enriched uranium assemblies), and IMF (Inert matrix fuel). The discussion begins with the direct disposal of spent LWR fuel for the reference Yucca Mountain operating conditions, and then compares the results with cases where the spent fuel has been processed and certain elements recycled to determine if there is a net benefit to such an approach, and to quantify the benefit. This is a continuation of previous efforts, [2,3], and uses the same detailed 3-D thermal model of the repository constructed using the SINDA/G computer code. This thermal model has been validated by

comparison to a number of published results for a repository at Yucca Mountain. As summarized in the previous section, the previous work had clearly identified the role of plutonium and americium in limiting the drift loading of a repository at Yucca Mountain, as well as the beneficial effects of subsequent removal of the fission product elements cesium and strontium for certain recycling scenarios.[2,3]

The analyses in this report all deal with cases where:

1. plutonium and americium are removed from spent fuel at 99.9% efficiency, and are recycled for further irradiation.
2. neptunium is also removed at 99.9% efficiency, and is recycled for a variety of reasons, including the possible potential to make the recycled fuel more diversion resistant and to reduce the radionuclide inventory of certain isotopes contributing to the peak dose rate.
3. cesium and strontium are separated with 100% efficiency from the waste stream, and are placed in separate storage.
4. curium and other actinides remain in the processing waste, along with the other fission products and are not recycled.

For each of these cases, detailed fabrication, separation, and irradiation histories had been calculated in a parallel effort by J.A. Stillman [4] and T.K. Kim [5] for a range of the number of recycles, with the assumption that all spent fuel was processed at 5 years after discharge to minimize buildup of americium from plutonium decay.

It is important to reemphasize that the benefits are being calculated for the current HTOM of a repository at Yucca Mountain. As mentioned above, the benefits may also apply if the temperature limits are lowered, although if the operating limits are changed too much, different impacts may result.[2] It should also be noted that the results are being calculated for water-moderated thermal systems, and the benefits should not necessarily be extrapolated to other thermal neutron systems, such as those with graphite moderation, without further study.

II. Direct Disposal of Commercial Spent Nuclear Fuel (CSNF)

The current reference case for a repository at Yucca Mountain is described in Ref. 1, and analysis results have been reported in Ref's 2 and 3. It has been assumed for these analyses that the spent fuel is placed at 25 years after discharge from the reactor, and that forced ventilation of the drifts is maintained for 75 years after waste placement (with the nominal planned ventilation rate of 15 m³/sec) to minimize the impact of shorter-lived isotopes. Also, only the spent PWR fuel component of CSNF is being considered, since the spent PWR fuel provides the highest heat sources in the repository, and constitutes the majority of the waste destined for disposal in a repository at Yucca Mountain. The spent PWR fuel is assumed to be loaded to the maximum possible drift loading consistent with satisfying the temperature limits. Such loading, and the associated temperatures, provide no margin for uncertainties, and do not necessarily correspond to the planned disposal operations for a repository at Yucca Mountain. The maximum allowable drift loading is used in this study only for the purposes of comparison between various approaches, although such comparisons are not greatly affected by incorporating margin for uncertainty. However, both the drift loading and the repository temperatures are highest when using the maximum allowable loading without including margins for uncertainty, and actual operating temperatures would be expected to be lower.

For the direct disposal of spent PWR fuel, subject to the assumptions listed in the previous paragraph, the maximum linear loading of a repository drift is approximately 1.15 MTIHM/m in order to satisfy the thermal constraints. For this reference case, the transient temperature behavior of a drift in the central region of the repository is shown in Fig. 1. Since this region is not near the repository boundary, this model minimizes effects associated with the finite length of the drifts, and the highest peak temperatures are expected. The decay heat is shown as a linear heat rate in the drift, W/m, and is shown decreasing in time consistent with the detailed decay heat data in Fig. 2.[4]

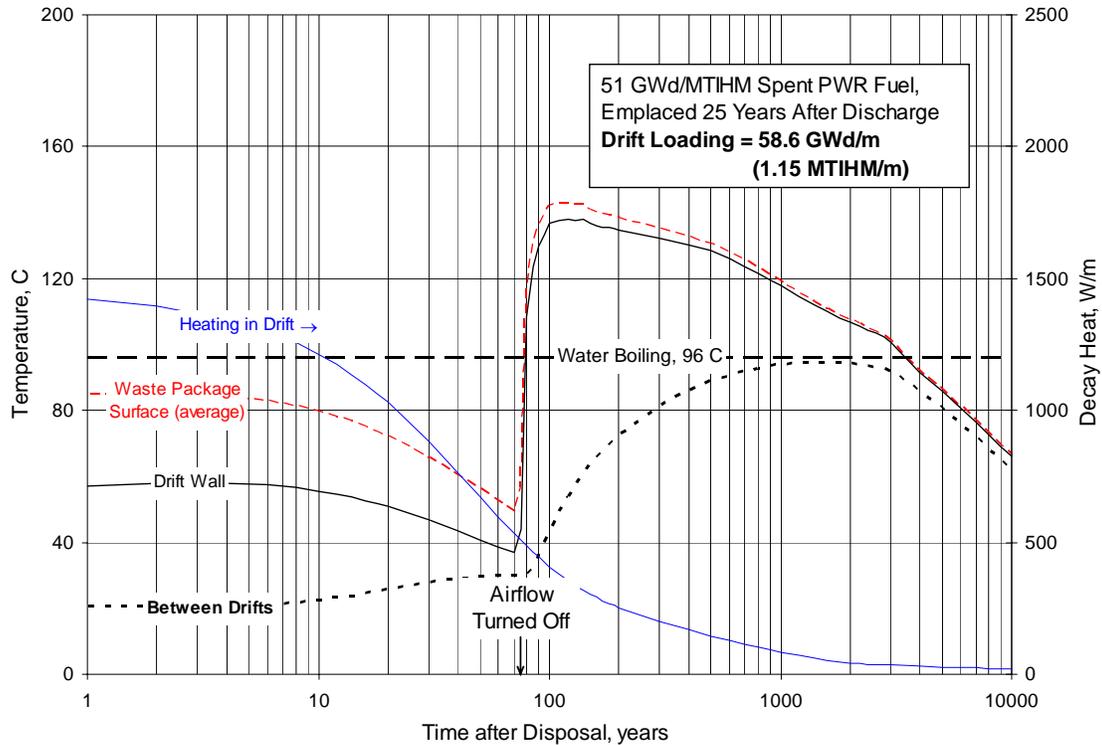


Figure 1. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for direct disposal of spent PWR fuel, 51 GWd/MTIHM discharge burnup.

The waste package surface temperature is initially below the boiling point of water while the drift is ventilated, but once forced ventilation is stopped at 75 years after disposal, the surface temperature increases rapidly to a peak of about 140 °C. The temperature then slowly drops over time, falling below boiling after about 3500 years. The drift wall temperature exhibits the same trends. The temperature midway between adjacent drifts responds much more slowly, since the main mechanism for heating the rock in this location is by conduction from the hotter areas around the drifts. As Fig. 1 shows, this temperature peaks just below 96 °C, one of the current temperature criteria for a repository at Yucca Mountain.

At this point, it should be noted that none of the temperatures are close to any of the other relevant temperature limits. For this reason, it has been concluded in previous work that the 96°C temperature limit midway between adjacent drifts is the controlling limit for the reference case, and that the peak temperature in this location occurs at about 1500-2000 years after waste placement.[2,3] Also, this temperature peak is mainly the result of the integrated decay heat over

time since repository closure, i.e., from 100 years after discharge from the reactor out to about 1500 years, rather than the value of decay heat at any particular time, since Fig.1 shows that substantial heating of the interior of the mountain does not commence until after forced ventilation has stopped. Therefore, the chemical element(s) in the spent PWR fuel causing the temperature limit to be approached would be those that contribute most to the integrated decay heat from the time when ventilation ceased until about 1500-2000 years.

Considering the decay heat data shown in Fig. 2, it is seen that the decay heat is dominated by actinide elements for all times past about 60 years, indicating that one or more of the actinide elements is responsible for the integrated decay heat causing the temperature peak midway between the drifts.[2] Examining the contributions from each of the actinide elements during the time period past 100 years out to 2000 years, it is observed that the dominant contribution is from Am-241. Since the isotopic composition of spent PWR fuel at the time of discharge shows little Am-241, the source of Am-241 is the radioactive decay of Pu-241, with a small contribution indirectly from the decay of Cm-245. Other important isotopes contributing to the decay heat during this time period are shown to be Pu-238, Pu-239, and Pu-240.

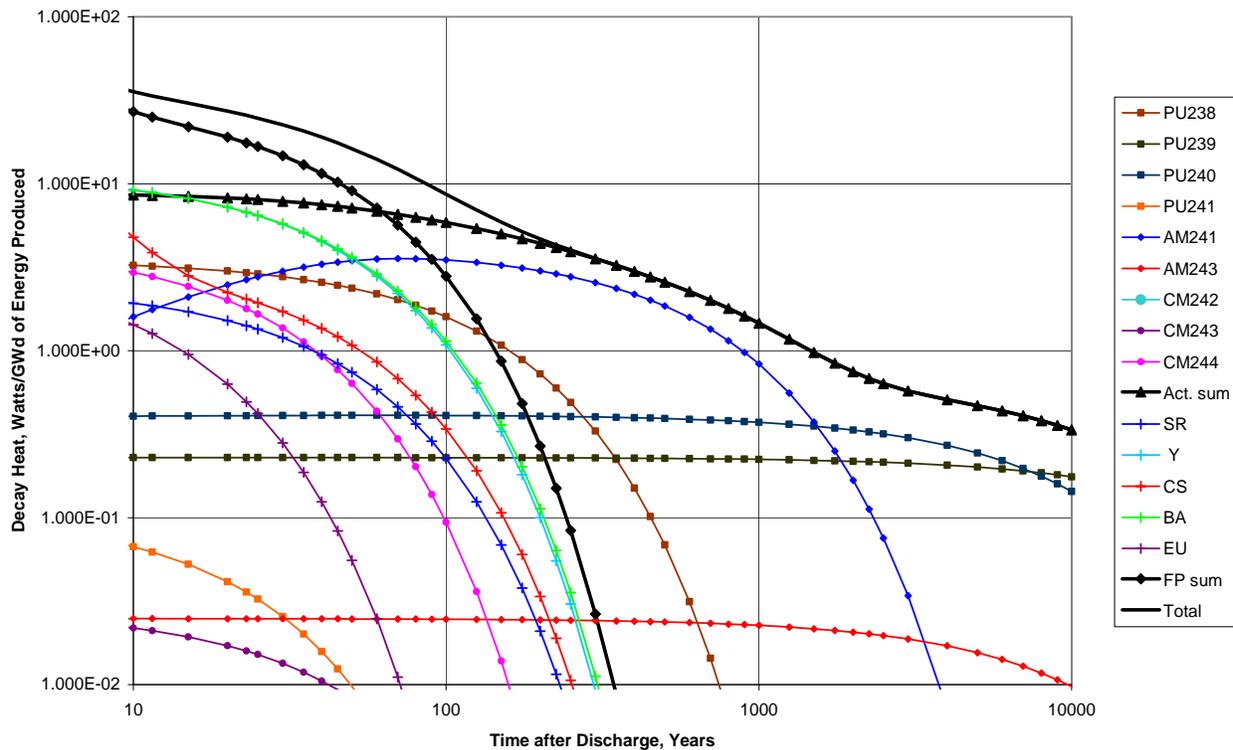


Figure 2. Decay heat characteristics of spent PWR fuel, 51 GWd/MTIHM discharge burnup.

As a result, it would appear that the current plans for the design and loading of a repository at Yucca Mountain are constrained by the integrated decay heat mainly from Am-241, which arises from decay of Pu-241, with the remainder coming from other plutonium isotopes. Therefore, to benefit the loading of the repository, it would be essential to remove the plutonium from the spent PWR fuel. In addition, the storage of spent fuel prior to processing allows buildup of Am-241, sufficient to require americium removal along with the plutonium. As stated in Section I.A, these results provide the justification for separating and recycling of plutonium and americium,

with the neptunium recycled for other purposes. Separation of cesium and strontium provide further benefit, at least for the nominal conditions assumed in this study, namely that placement in the repository occurs 25 years after discharge from the reactor, and that ventilation of the repository is sustained for 75 years after placement of the waste.

III. Recycling of Plutonium, Americium, and Neptunium as MOX

The use of MOX (mixed-oxide) fuel in LWRs has been proposed as a possible approach to provide benefits to a geologic repository.[6] In this study, the MOX fuel is fabricated from plutonium, americium, and neptunium, in a matrix of the recovered uranium. Spent PWR fuel is used as the source of plutonium, americium, and neptunium for the MOX fuel. The MOX assembly characteristics in each of the following cases are from lattice calculations, as described below. The PWR MOX strategy uses the recovered materials from a number of processed PWR assemblies to provide sufficient fissile content for a single MOX assembly. Subsequent recycling of MOX assemblies also requires the use of a number of MOX assemblies from the previous generation to provide enough fissile material for the next generation of MOX fuel. Each MOX assembly is assumed to produce the same integrated energy as the original PWR assembly, so that the MOX assemblies can be viewed as substitutes for additional PWR assemblies. It is assumed that the irradiated MOX assemblies in the last generation are directly disposed in the repository, without processing. Therefore, for each case, the spent MOX assemblies from the last irradiation are disposed directly into the repository, after an appropriate cooling time, along with the process waste from the spent PWR fuel and all previous recycled MOX assemblies.[4]

In this study, the PWR MOX strategy uses the following assumptions.

1. The spent PWR fuel that is processed to remove the plutonium, americium, and neptunium is assumed to have decayed for 5 years after discharge from the reactor to minimize the creation of Am-241 and to maximize the potential benefit for increasing the repository drift loading.
2. The processing is assumed to be capable of removing 99.9% of the plutonium, americium, and neptunium from the spent PWR fuel, i.e., almost all of the plutonium, americium, and neptunium are recovered for use in fabricating the MOX fuel.
3. The fission products cesium and strontium are removed from the resulting processing waste and are stored separately. Curium is not recycled, and stays in the process waste with all of the remaining fission products.
4. After fabrication, there is a delay of 2 years before the MOX assemblies are loaded into the reactor for irradiation, so that each MOX recycle requires 11.5 years.
5. The neutronic analyses for the enriched uranium and MOX fuel are performed using lattice calculations. This is roughly equivalent to treating the assemblies as if they were in reactor cores of identical assemblies, i.e., homogeneous reactor cores. In some cases, the use of homogeneous reactor cores leads to difficulties with reactor safety coefficients and other issues that may result in these cases being impractical, requiring the use of heterogeneous reactor cores.[4] However, based on previous studies, it is expected that the results of the lattice calculations for spent fuel compositions would be very close to the results obtained for heterogeneous reactor cores.

Table 1. Assembly requirements for fabricating each generation of MOX assembly [4]

		TRU	Assemblies to process to fabricate one assembly in next generation	Total Assemblies				
				N=1	N=2	N=3	N=4	N=5
Once-through UO ₂	Charged	0.0						
	Reprocessed	6.0	13.50	13.50	26.74	40.66	54.75	68.97
	Recycled	5.9		(93.1%)	(90.0%)	(88.0%)	(86.6%)	(85.6%)
1st Recycle MOX	Fabricated	80.3						
	Charged	80.3						
	Reprocessed	69.2	1.98	1.0	1.98	3.01	4.06	5.11
	Recycled	68.2		(6.9%)	(6.7%)	(6.5%)	(6.4%)	(6.3%)
2nd Recycle MOX	Fabricated	135.2						
	Charged	135.1						
	Reprocessed	121.1	1.52		1.0	1.52	2.05	2.58
	Recycled	119.5			(3.4%)	(3.3%)	(3.2%)	(3.2%)
3rd Recycle MOX	Fabricated	181.8						
	Charged	181.6						
	Reprocessed	165.4	1.35			1.0	1.35	1.70
	Recycled	163.4				(2.2%)	(2.1%)	(2.1%)
4th Recycle MOX	Fabricated	220.0						
	Charged	219.8						
	Reprocessed	201.9	1.26				1.0	1.26
	Recycled	199.6					(1.6%)	(1.6%)
5th Recycle MOX	Fabricated	251.4						
	Charged	251.0						
	Reprocessed	231.8	N/A					1.0
	Recycled	229.2						(1.2%)

To provide a basis for comparison between cases with recycling and the reference case of direct disposal of spent PWR fuel, groups of assemblies are formed for each recycle case, with each assembly in each generation irradiated to provide the same integrated energy. For example, in the case of the first recycle MOX, designated ‘MOX-1’, 13.5 spent PWR assemblies are required to create one new MOX assembly. The group of the 13.5 PWR assemblies and the 1 MOX-1 assembly constitute the MOX-1 group, with a total of 14.5 assemblies. The characteristics of the MOX-1 group are therefore represented by the waste from processing the 13.5 spent PWR assemblies, removal of the cesium and strontium from the waste, and direct disposal of the spent MOX-1 assembly. The characteristics of the MOX-1 group are then compared to a group of spent PWR assemblies consisting of 14.5 assemblies, all directly disposed. In this way, **each group represents the same total integrated energy generation.** (Due to the fabricated MOX fuel characteristics, this causes the MTIHM between the two groups to be virtually the same, and allows results to be considered on either a MTIHM or GWd basis.)

III.A MOX-1

As described above, the MOX-1 case is created by processing spent PWR fuel, and recycling the plutonium, americium, and neptunium along with the recovered uranium to form MOX fuel. The content of the MOX-1 group is as follows:

- 1 spent MOX-1 fuel assembly, direct disposal

- waste from processing 13.5 spent PWR fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- total material in the MOX-1 group is from 14.5 fuel assemblies

The MOX fuel is irradiated in a light-water reactor to the same integrated energy as a PWR assembly. The spent MOX assembly and the process waste are then all placed in the repository 25 years after discharge from the reactor. The transient temperature behavior for the MOX-1 group in the repository is shown in Figure 3.

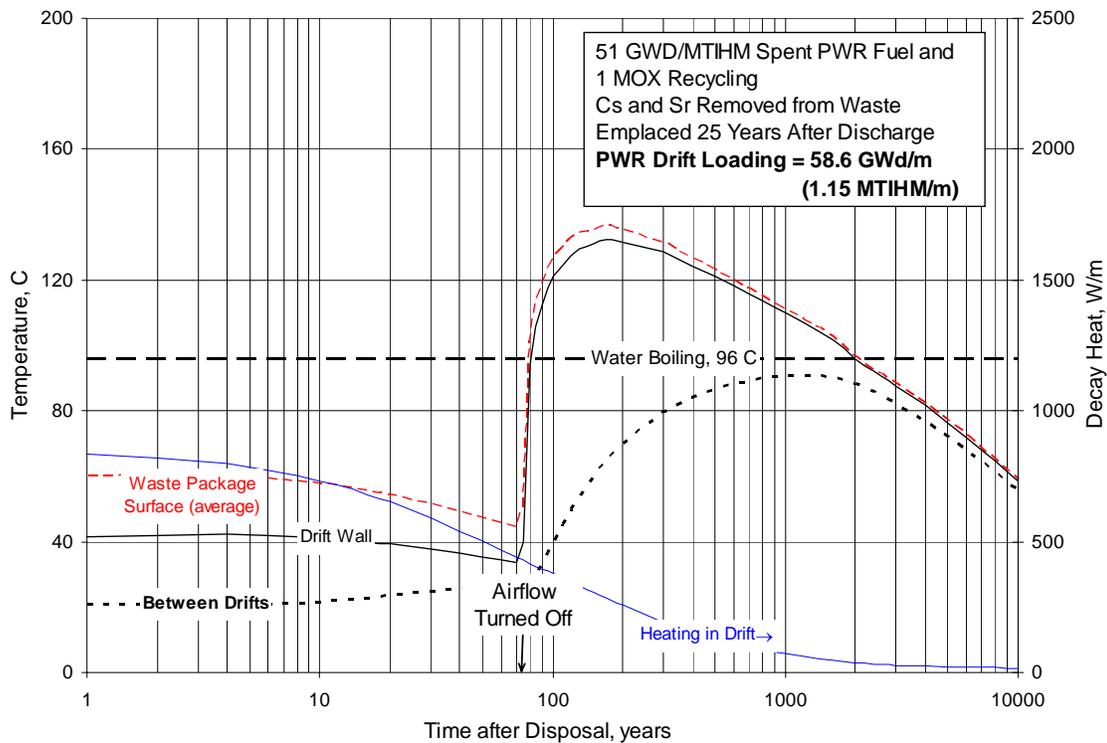


Figure 3. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the MOX-1 group, assemblies irradiated to 51 GWd/MTIHM, with the reference drift loading of 1.15 MTIHM/m.

As shown in the figure, the transient thermal response is almost identical to that for the direct disposal of spent PWR fuel. Although it is not readily apparent, there has been a drop in the peak temperature midway between the drifts to 91°C. This small drop in peak temperature provides the opportunity to increase the maximum drift loading, as shown in Fig. 4. The maximum allowable drift loading has been increased from 1.15 MTIHM/m (representing material that produced 58.6 GWd of energy being stored per meter of drift) to 1.25 MTIHM/m (representing material that produced 63.8 GWd of energy being stored per meter of drift), or an increase by a factor of almost 1.09. The limiting temperature in the repository for the MOX-1 case is still the peak temperature midway between the drifts, with the peak temperature occurring around 1500 years after placement.

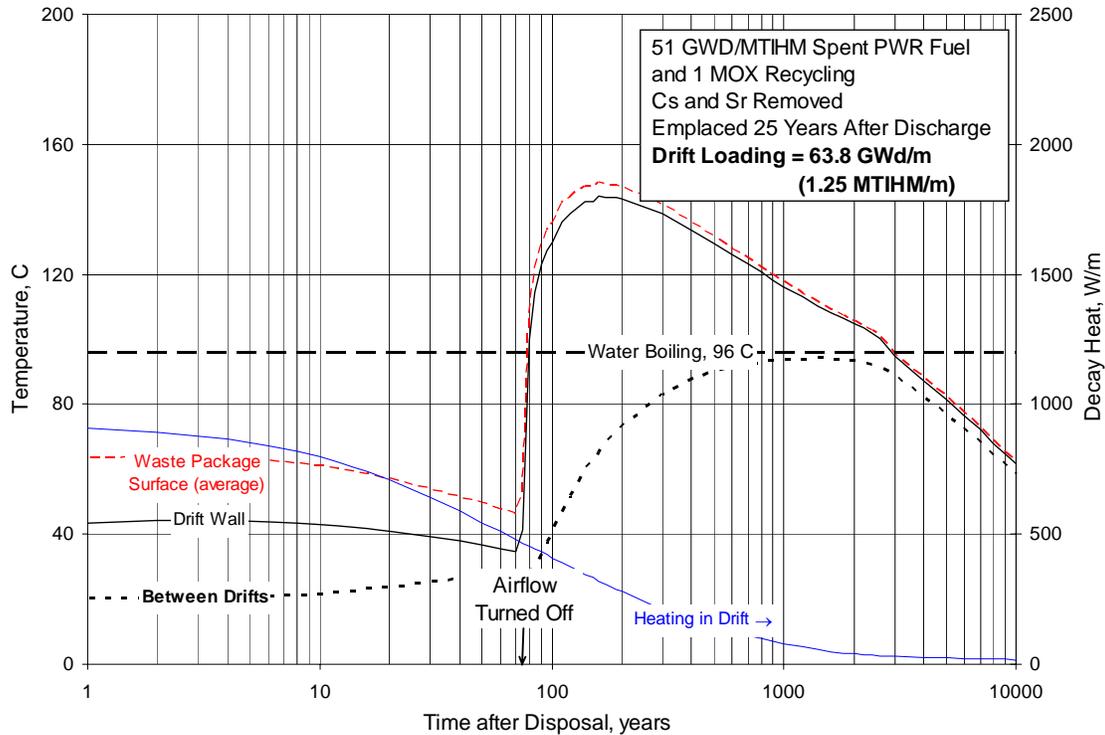


Figure 4. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the MOX-1 group, assemblies irradiated to 51 GWd/MTIHM; loading increased to the maximum drift loading of 1.25 MTIHM/m.

The isotopic contributions to the decay heat in the MOX-1 group are generally similar to those for the direct disposal of spent PWR fuel, with some important differences, as shown in Figure 5. There is a decrease in the contribution from Am-241, consistent with the lower discharge amount of Pu-241. There is an increase in the Pu-238 decay heat, created partly from capture in Np-237, and partly from an increase in the Cm-242 content (arising from americium), with each path contributing about equally to the increase. The overall effect on the MOX-1 group decay heat is: a reduction in decay heat from discharge out to about 150 years, very little change in the time period from 150 years to 350 years, and a reduction after that time. The integrated decay heat for the MOX-1 group is therefore lower than for the direct disposal PWR group (per unit energy produced), and accounts for the allowable increase in drift loading. It is interesting to note that the increase in the Pu-238 content offsets some of the gains made from the lower Pu-241 content during this time period, and appears to reduce the benefit that might be expected from the first MOX irradiation. The effect of the increased Pu-238 content can be partially mitigated by not recycling the neptunium, but the potential drift loading increase from this strategy has not been evaluated at this time, and would be the subject of further analyses.

To determine the importance of removing the cesium and strontium fission products, the MOX-1 group was also analyzed without the removal of the cesium and strontium fission products from the process waste. The thermal response is shown in Figure 6, where it is observed that the controlling temperature limit is still the long-term temperature peak midway between adjacent drifts, although the temperature peak immediately after closure is also higher. The maximum drift loading is now 1.20 MTIHM/m, reducing the benefit of the MOX-1 group by about 50%.

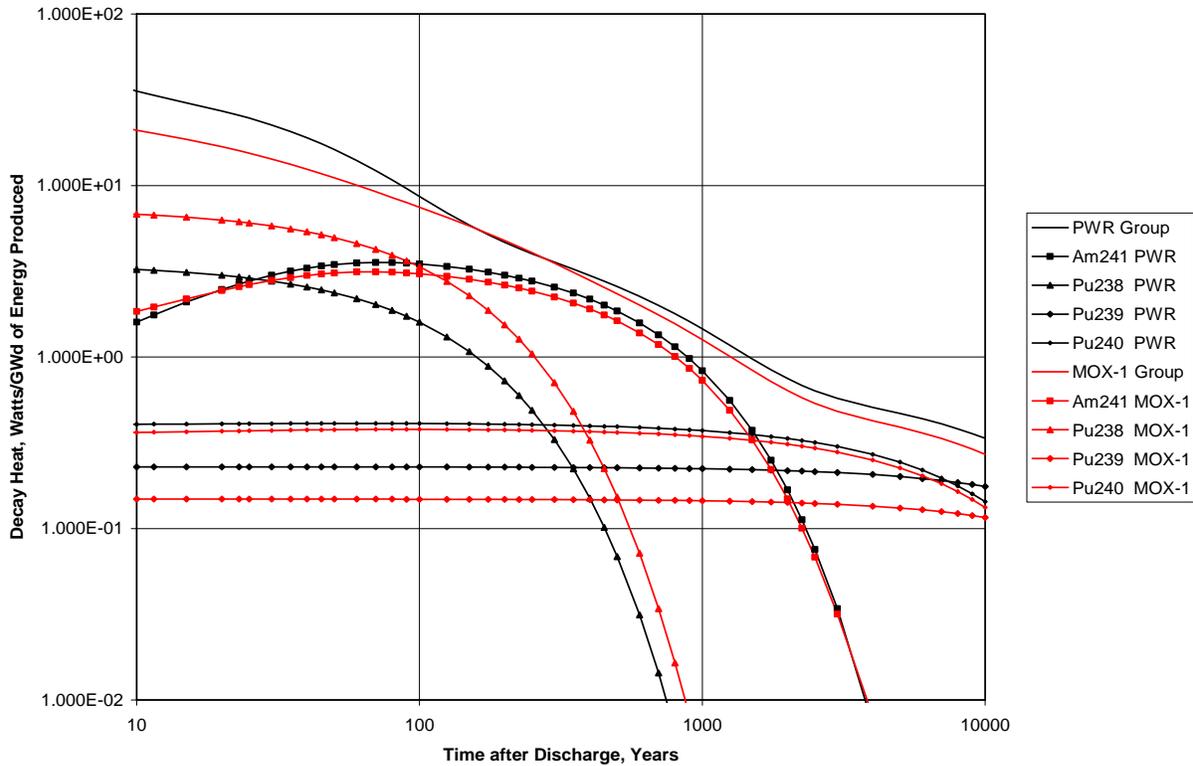


Figure 5. Decay heat characteristics for the MOX-1 group compared to the spent PWR fuel group, assemblies irradiated to 51 GWd/MTIHM discharge burnup.

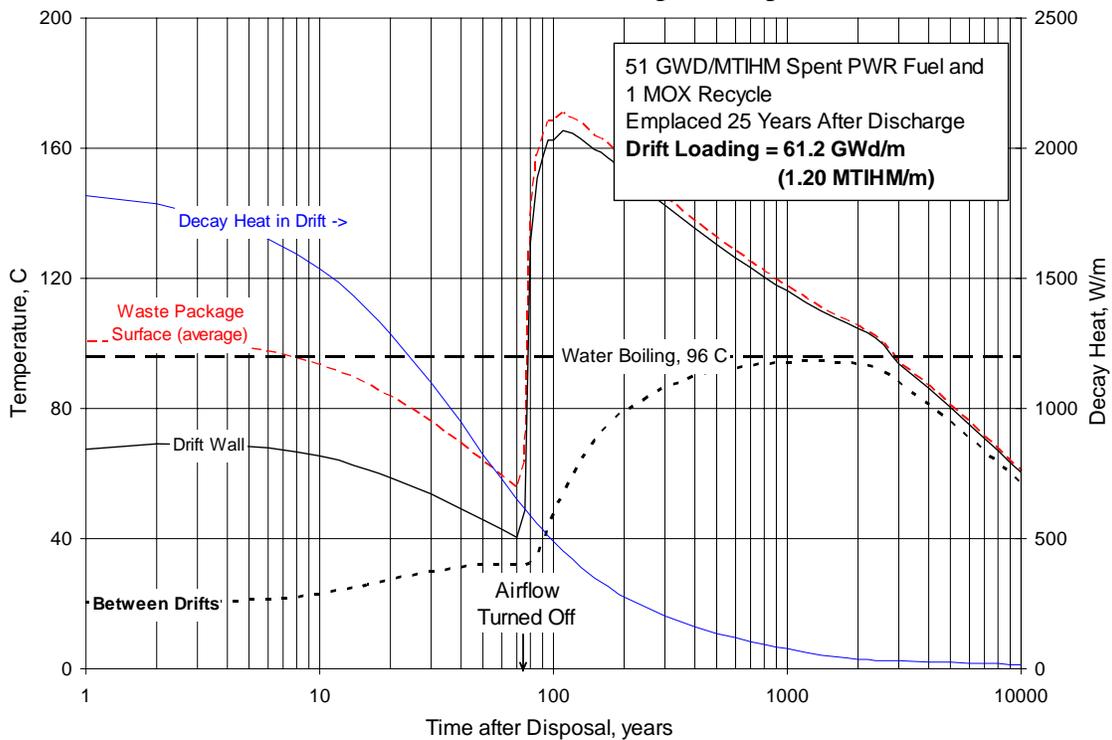


Figure 6. Transient temperature response of drifts in the central region of a repository at Yucca Mountain for the MOX-1 group, without removal of the cesium and strontium from the processing waste, assemblies irradiated to 51 GWd/MTIHM; increased drift loading.

One could also consider processing the single spent MOX-1 fuel assembly in the group to remove the cesium and strontium prior to disposal, similar to what is done for the spent PWR fuel process waste. Given that removing the cesium and strontium from the 13.5 spent PWR fuel assemblies that were processed to make one MOX assembly approximately doubled the benefit of using MOX, subsequent removal from one more assembly would be expected to be of small benefit. Preliminary estimates are that the additional benefit is barely noticeable. It should also be noted that the magnitude of the effect of removing cesium and strontium also depends on the cooling time prior to placement in the repository, and on the length of the ventilation period.

III.B MOX-2

The MOX-2 group is created by taking the discharged MOX assemblies from the MOX-1 case, processing them to remove the plutonium, americium, and neptunium, and fabricating new MOX-2 assemblies. As listed in Table 1, the fabrication of one of the MOX-2 assemblies, capable of irradiation to 51 GWd/MTIHM, requires processing 1.98 MOX-1 assemblies (which in turn would require the processing of 26.74 spent PWR assemblies). The MOX-2 group therefore consists of:

- 1 spent MOX-2 fuel assembly, direct disposal
- waste from processing 1.98 spent MOX-1 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 26.74 spent PWR fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- total material in the MOX-2 group is from 29.72 fuel assemblies

The performance of this group in the repository is compared to a group of 29.72 directly-disposed spent PWR fuel assemblies, preserving the total integrated energy generation between the two groups. It is emphasized that cesium and strontium are removed from the processing waste, and the single MOX-2 assembly in the group is directly-disposed without processing. (As in the previous case, removal of the cesium and strontium from the one MOX-2 assembly would make very little difference in the decay heat for the group of 29.72 assemblies.)

The MOX-2 group has lower decay heat than the MOX-1 group and spent PWR fuel, allowing the maximum drift loading to be further increased up to 1.36 MTIHM/m (representing material that produced 69.4 GWd of energy being stored per meter of drift) to achieve the same peak temperature midway between adjacent drifts as in the reference case. The transient temperature response of the repository is shown in Figure 7, with the repository response to the MOX-2 group at increased drift loading being almost identical to the response for the direct disposal of spent PWR fuel as shown in Figure 1.

Comparing the contributions to the total decay heat from the dominant contributors, the Am-241 contribution is further reduced from the MOX-1 case, while there is another small increase in the Pu-238 contribution, as shown in Figure 8. This continues the trend observed between the MOX-1 case and the direct disposal of spent PWR fuel. The net effect of the different isotopic distribution for the MOX-2 group is to provide a reduction in decay heat out to about 150 years, no change in the decay heat from about 150 years out to 300 years, and lower decay heat after that time. As a result, the integrated decay heat from 100 years after discharge out to about 1500 years is lower in the MOX-2 group than in the MOX-1 group (per unit energy produced), and allows the further increase in drift loading with the MOX-2 case. The question of the impact of

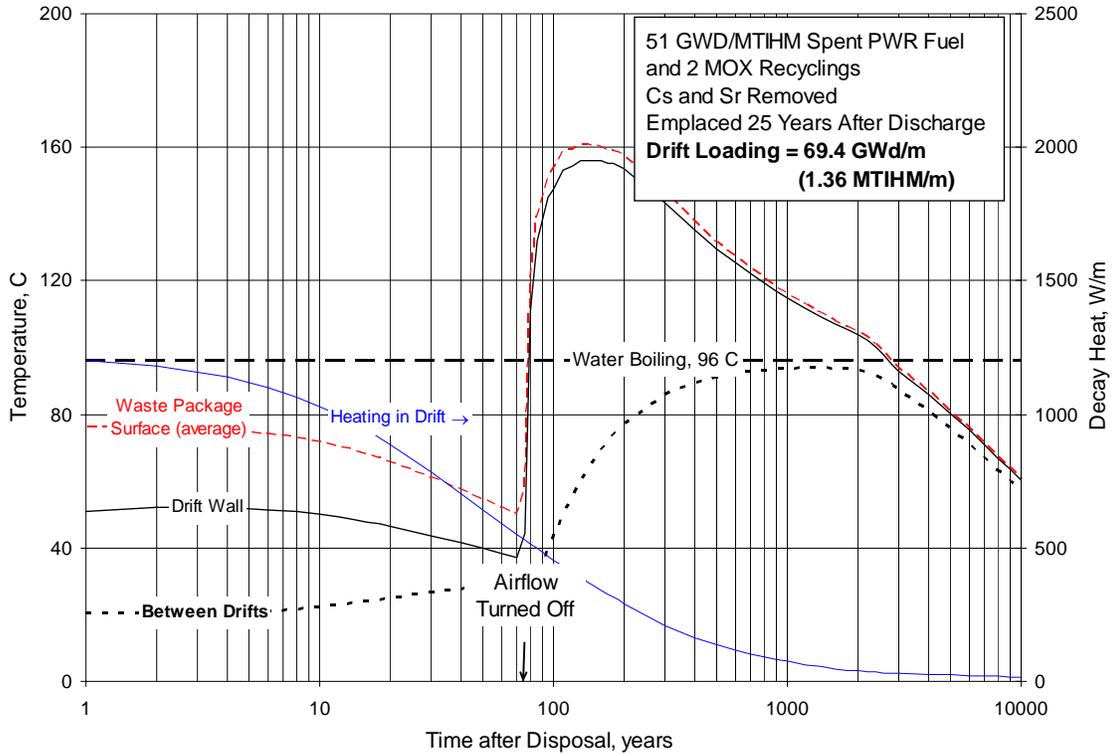


Figure 7. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the MOX-2 group, assemblies irradiated to 51 GWd/MTIHM; loading increased to a maximum drift loading of 1.36 MTIHM/m.

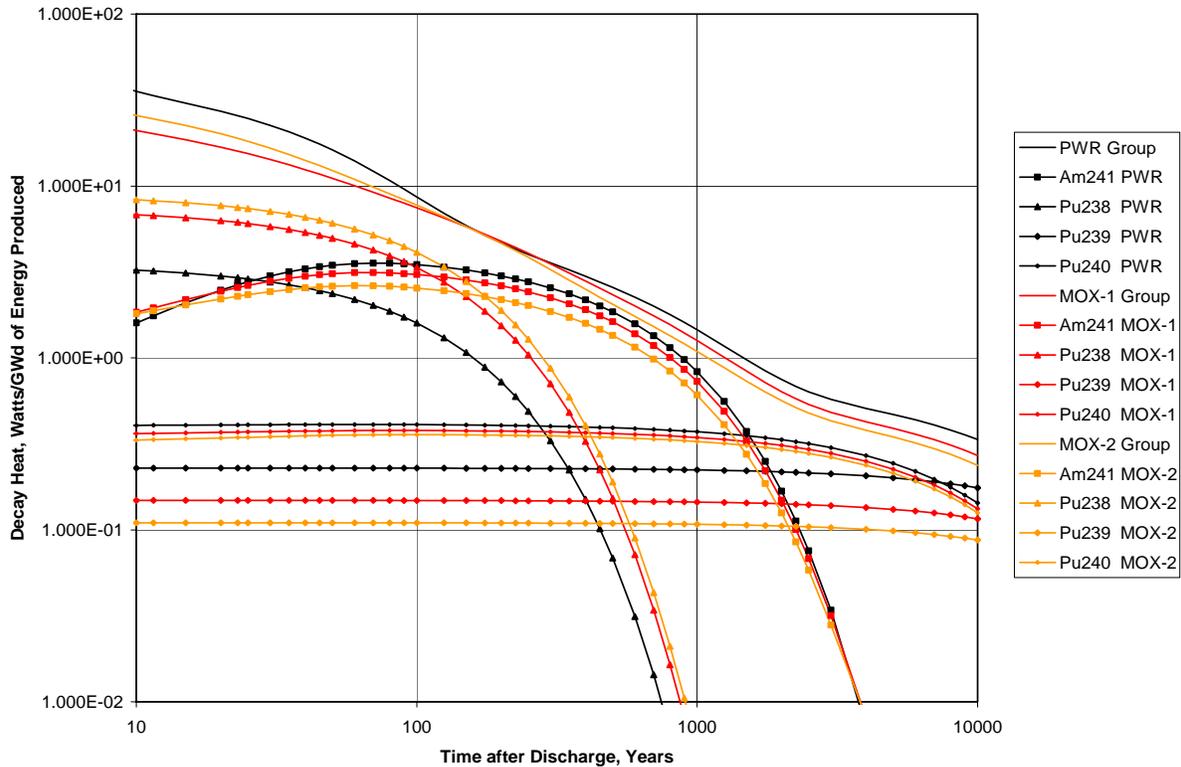


Figure 8. Decay heat characteristics for the MOX-2 group, compared to the MOX-1 group and direct disposal of spent PWR fuel, all assemblies irradiated to 51 GWd/MTIHM.

neptunium recycling on contributing to the increase in the Pu-238 content in the MOX-2 group still needs to be addressed in future work, as with the MOX-1 group. It should be noted that there is essentially no change in the decay heat contribution from Pu-240, although each MOX recycle results in some reduction in the contribution from Pu-239.

III.C MOX-3

Creating the grouping for the third recycle of MOX fuel, MOX-3, in the same fashion as for the MOX-1 and MOX-2 groups, referring to Table 1, results in the following group for MOX-3:

- 1 spent MOX-3 fuel assembly, direct disposal
- waste from processing 1.52 spent MOX-2 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 3.01 spent MOX-1 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 40.66 spent PWR fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- total material in the MOX-3 group is from 46.19 assemblies

This group is compared to a group of 46.19 spent PWR fuel assemblies, to preserve the total integrated energy. The transient behavior of the repository in response to the MOX-3 group is very much like the MOX-1 and MOX-2 cases, although the maximum drift loading can be further increased to 1.46 MTIHM/m, representing material that produced 74.5 GWd of energy being stored per meter of drift, while still satisfying the temperature limit of 96 °C between adjacent drifts. The contributions to the decay heat from the various isotopes continue the trend observed for the MOX-1 and MOX-2 groups, accounting for the small decrease in integrated decay heat generation in the MOX-3 group (on a per unit energy basis).

III.D MOX-4

Continuing the recycling, the MOX-4 assemblies are created by processing the MOX-3 assemblies and recovering the plutonium, americium, and neptunium. Cesium and strontium are removed from all process waste. The composition of the MOX-4 group is as follows, according to Table 1:

- 1 spent MOX-4 fuel assembly, direct disposed
- waste from processing 1.35 spent MOX-3 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 2.05 spent MOX-2 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 4.06 spent MOX-1 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 54.75 spent PWR fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- total material in the MOX-4 group is from 63.21 assemblies

All assemblies are irradiated to provide the same integrated energy. This group is then compared to the direct disposal of 63.21 assemblies of spent PWR fuel, preserving the total integrated energy. The transient repository response for disposal of the MOX-4 group is again quite similar to the previous MOX cases. The maximum drift loading can be increased to 1.60 MTIHM/m,

representing material that produced 81.6 GWd of energy being stored per meter of drift, while still satisfying the temperature limit of 96 °C between adjacent drifts. The change in the isotopic content of the disposed materials continues the trends already noted, causing the decay heat to be lower for the MOX-4 group than for the previous MOX groups and spent PWR fuel.

III.E MOX-5

The MOX-5 assemblies are created by processing the MOX-4 assemblies and recovering the plutonium, americium, and neptunium. The MOX-5 group is then formed from the following assemblies, according to Table 1:

- 1 spent MOX-5 fuel assembly, direct disposal
- waste from processing 1.26 spent MOX-4 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 1.70 spent MOX-3 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 2.58 spent MOX-2 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 5.11 spent MOX-1 fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- waste from processing 68.97 spent PWR fuel assemblies, 99.9% of plutonium, americium, and neptunium removed, cesium and strontium removed
- total material in the MOX-5 group is from 80.6 assemblies

All assemblies in the MOX-5 group were irradiated to provide the same integrated energy from each assembly. The MOX-5 group is compared to a similar group of 80.6 spent PWR fuel assemblies, preserving the total energy generation. The transient temperature behavior for the MOX-5 group in the repository is shown in Figure 9 for the same drift loading of 1.15 MTIHM as direct disposal of spent PWR fuel.

As is shown in the graph, the peak temperature is now much lower than for the spent PWR fuel group, reaching a maximum of about 72.5 °C as a result of the lower decay heat for the MOX-5 group. This allows for a substantial increase in maximum drift loading up to 1.71 MTIHM/m, representing material that produced 87.2 GWd of energy being stored per meter of drift. The transient behavior of the MOX-5 group in the repository with the increased drift loading is shown in Figure 10, where the overall response is similar to the previous MOX cases and to the case for direct disposal of spent PWR fuel, although the increased temperature immediately after the forced ventilation is stopped should be noted. As with all of the previous cases, the decay heat for the MOX-5 group is lower due to the continuing changes in the isotopic content of the disposed material, following the trends discussed earlier.

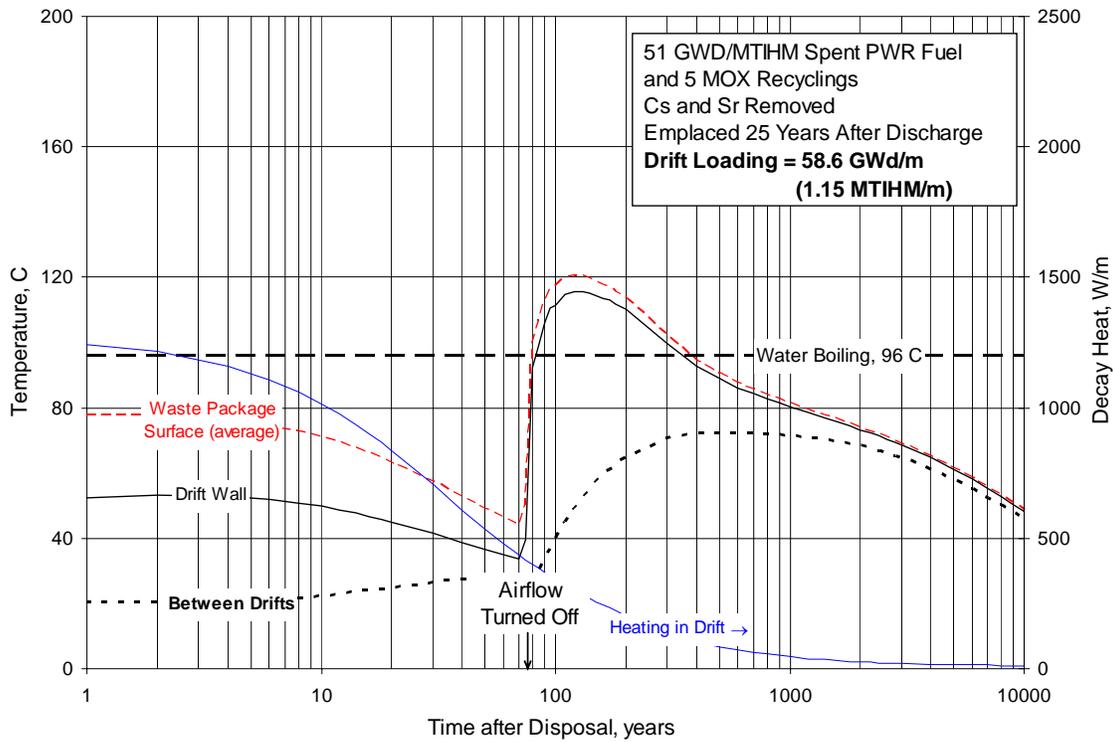


Figure 9. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the MOX-5 group, assemblies irradiated to 51 GWd/MTIHM; reference drift loading of 1.15 MTIHM/m

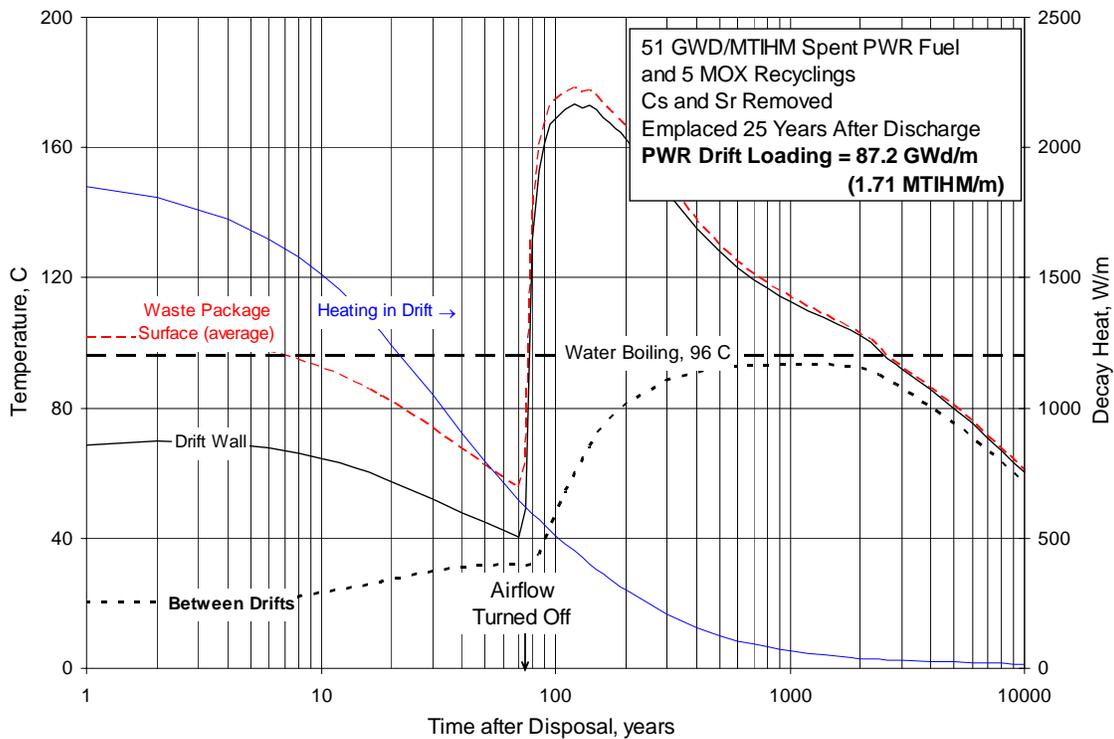


Figure 10. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the MOX-5 group, assemblies irradiated to 51 GWd/MTIHM; loading increased to a maximum drift loading of 1.71 MTIHM/m

III.F Summary of MOX Results

The repository benefit of using repeated recycle of MOX assemblies is shown in Figure 11, where the increase in drift loading (GWd/m) relative to the reference case is plotted as a function of the number of recycling steps. With each recycle, there is a slow but steady increase in the drift loading, consistent with the reduced plutonium inventory at discharge compared to using once-through PWR fuel. The effect of leaving the cesium and strontium fission products in the waste is also included on the graph. Considering the changes to the isotopic distribution, it is observed that both the Pu-239 and Pu-241 content drop with each recycle step, following a general trend of consuming plutonium. However, there is a notable increase in the content of the higher isotopes, both of plutonium and of the higher actinides in general. There is also a steady increase in Pu-238 content with each recycle, coming from both the higher Np-237 content at each recycle step, as well as the higher Cm-242 content, reducing the benefit that would otherwise accrue from reducing the Pu-241 content (given that the dominant decay heat contributor is still Am-241, arising almost entirely from the decay of Pu-241 after discharge of the spent fuel from the reactor). The increase in the higher actinide content appears to be the result of increasing concentrations of plutonium and americium isotopes in the charge at each step through reliance on these elements for the fissile content, causing creation of more of the higher isotopes, although a detailed investigation of the isotopic mass flows in each group still needs to be performed to fully understand and explain the results.

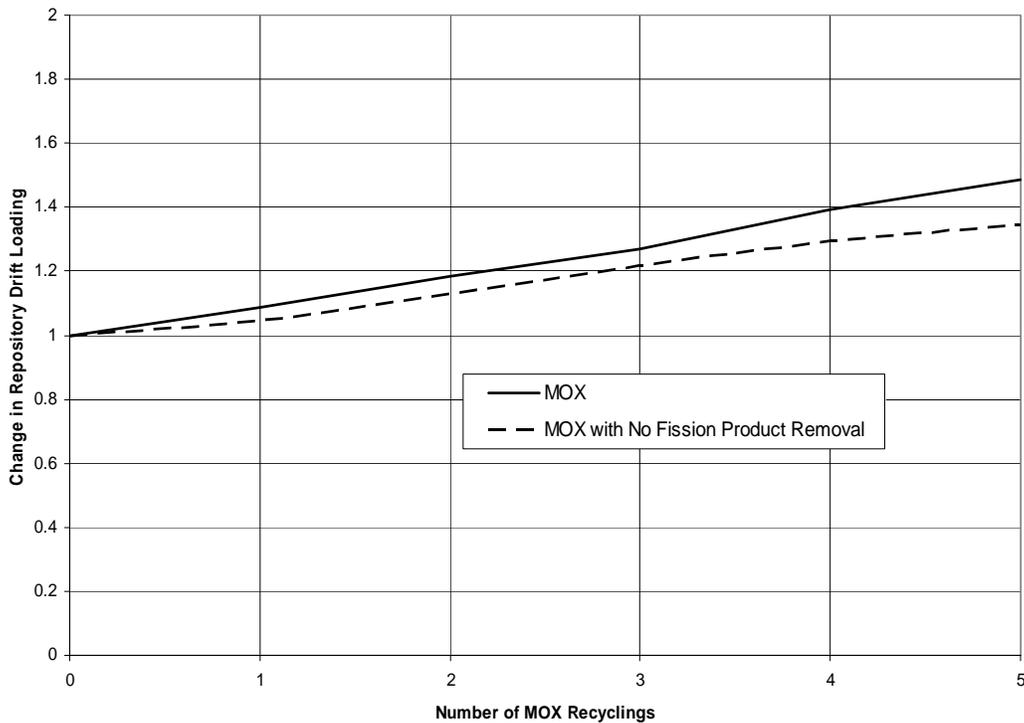


Figure 11. Impact of using MOX recycling on drift loading of a repository at Yucca Mountain with and without the removal of cesium and strontium from the process waste

However, before moving on to the other recycling strategies, there is one additional observation that needs to be discussed. While the repository loading is limited by the peak temperature occurring midway between adjacent drifts for all of the MOX recycle cases currently analyzed,

the peak temperature immediately after the closure of the repository is steadily increasing with each recycle, see Fig. 10 for the example after 5 recycles with MOX. At some point, with enough recycle steps, some of the other temperature limits may become the controlling ones, and the benefits of further processing and recycling in the same manner will be much different. As a consequence, the trend presented in Fig. 11 should not be extrapolated too far beyond the current number of recycle steps.

IV. Recycling of Plutonium, Americium, and Neptunium Using the CORAIL Concept

A different approach to the recycling of the plutonium, americium, and neptunium actinide elements is to use heterogeneous fuel assemblies, where some of the fuel pins are fabricated from reprocessed materials, and the remainder is fabricated from enriched uranium. This is referred to as the CORAIL concept. The specific example used here is designated as CORAIL-PNA, to reflect the recycling of plutonium, neptunium, and americium. The principle used in this study is to create one heterogeneous fuel assembly from each discharged spent PWR fuel assembly, with enriched uranium added to provide sufficient fissile material, rather than the MOX approach of using the plutonium, americium, and neptunium from many spent PWR fuel assemblies to provide the total fissile content of a single assembly, with no contribution from enriched uranium. In the examples presented here, the fuel pins fabricated from reprocessed material constitute about 32% of the total number of fuel pins, with the remaining 68% being new enriched uranium fuel pins.

For multiple recycling using the heterogeneous assembly concept, the entire assembly is processed to recover the plutonium, americium, and neptunium from both the irradiated enriched uranium pins and the irradiated pins that were fabricated from reprocessed material. The recovered materials are then used to form new fuel pins to be used in the next irradiation step. These fuel pins are also supplemented with new enriched uranium fuel pins to create the next generation CORAIL-PNA assembly. The uranium enrichment is increased with each generation as fertile isotopes of the higher actinides build up. All of the neutronic results for this section were provided by T.K. Kim. [5]

IV.A Direct Disposal of Spent PWR Fuel at 45 GWd/MTIHM

The CORAIL-PNA assembly cases are all analyzed using a discharge burnup of 45 GWd/MTIHM, somewhat lower than the 51 GWd/MTIHM used for the MOX cases. For this reason, another reference case for the direct disposal of spent PWR fuel is shown in Figure 12 with this discharge burnup, showing the slightly higher maximum drift loading of 1.28 MTIHM/m due to the lower burnup, although this represents material that produced only 57.6 GWd of energy being stored per meter, slightly lower than the result obtained at the higher burnup. Apparently, this level of burnup is already into the region of irradiation where increasing burnup of spent PWR fuel has a repository benefit in reducing some of the plutonium isotopes, due to the increasing dependence on plutonium for energy production as irradiation continues to higher burnup. Since the previous analyses showed that the actinide elements are responsible for limiting disposal in the repository, the plot of decay heat for the actinide elements is shown in Figure 13 for comparison with Figure 2. The results are quite similar to those for the 51 GWd/MTIHM burnup case.

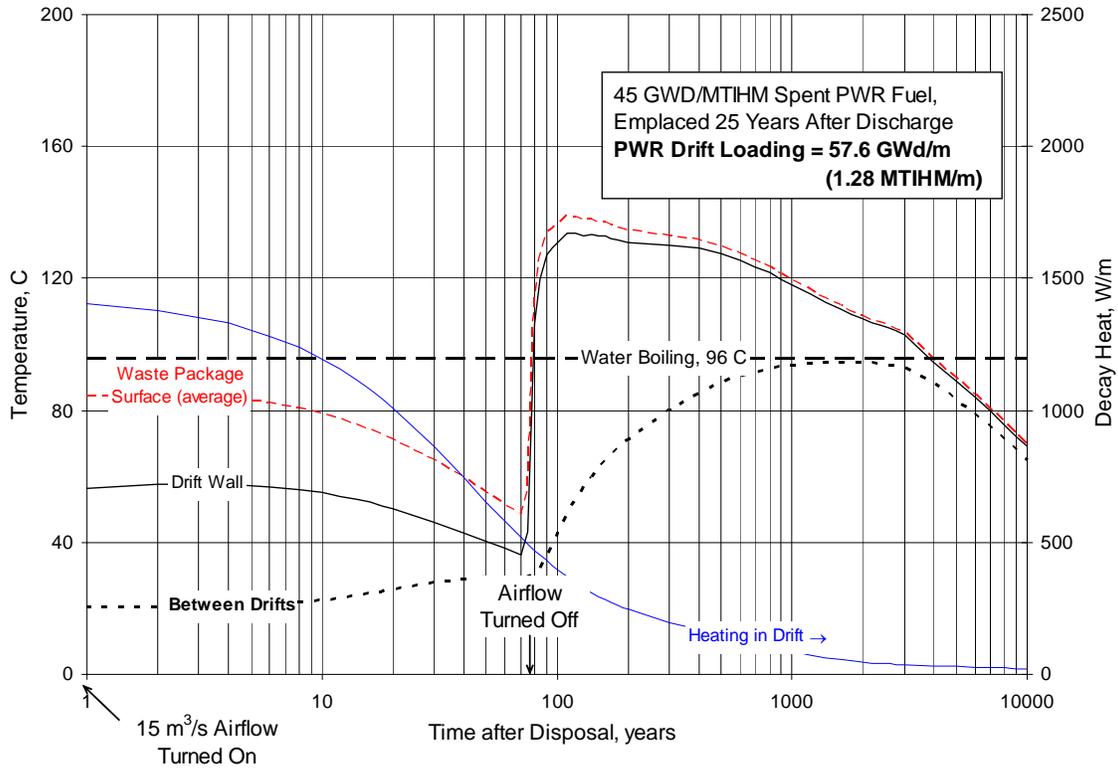


Figure 12. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for direct disposal of spent PWR fuel, 45 GWd/MTIHM discharge burnup; loading at the maximum drift loading of 1.28 MTIHM/m.

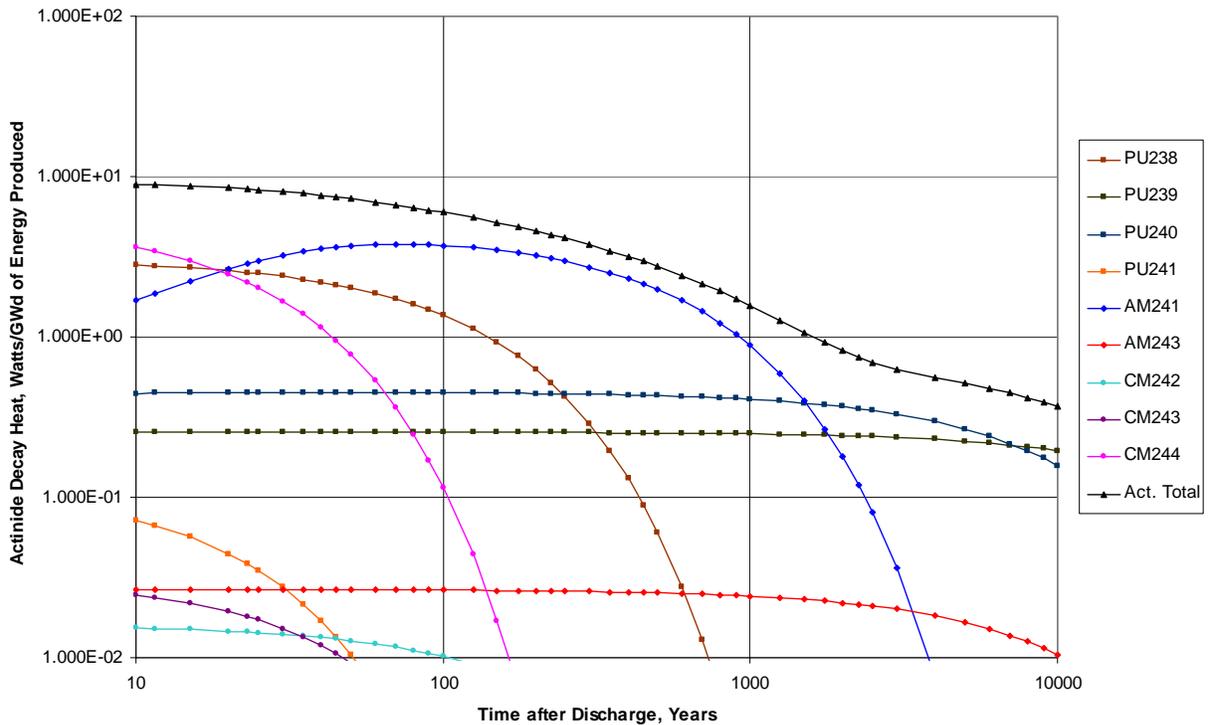


Figure 13. Actinide decay heat characteristics of spent PWR fuel, 45 GWd/MTIHM discharge burnup.

IV.B CORAIL-PNA-1

Each assembly in the first CORAIL-PNA pass, CORAIL-PNA-1, contains about 32% of the fuel pins that are fabricated from the plutonium, americium, and neptunium recovered from the spent PWR fuel assembly, in a matrix of depleted uranium. The rest of the pins in the assembly are fabricated from enriched uranium, with enrichment higher than that of the original PWR core, 4.87% as compared to 3.82%. However, the plutonium, americium, and neptunium content of the CORAIL-PNA-1 assembly are much less than for a MOX-1 assembly. For comparison with the direct disposal of spent PWR fuel, the CORAIL-PNA-1 assembly is grouped with the processing waste from the PWR assembly, minus the cesium and strontium, representing material from 2 assemblies. As in the MOX cases, the curium is not separated, but remains with the process waste. The repository behavior for the CORAIL-PNA-1 group is compared with the behavior for the direct disposal of a group of 2 PWR assemblies, with all assemblies being irradiated to 45 GWd/MTIHM, so that total energy generation is the same for the two groups.

The transient temperature behavior of the repository for the CORAIL-PNA-1 group is given in Figure 14. The drift loading has been maintained at the reference value of 1.28 MTIHM/m. While the overall transient behavior is similar, the peak temperature is lower at about 89 °C. This allows the drift loading to be increased up to 1.50 MTIHM/m, (representing material that produced 67.5 GWd of energy being stored per meter of drift), or rather the waste from processing this quantity of spent fuel being stored per meter of drift. This is a factor of 1.17 over the direct disposal of spent PWR fuel, which is much greater than the benefit observed with the first MOX recycle. The results for the increased loading are shown in Figure 15, where it is observed that the overall transient behavior is very similar to the reference case.

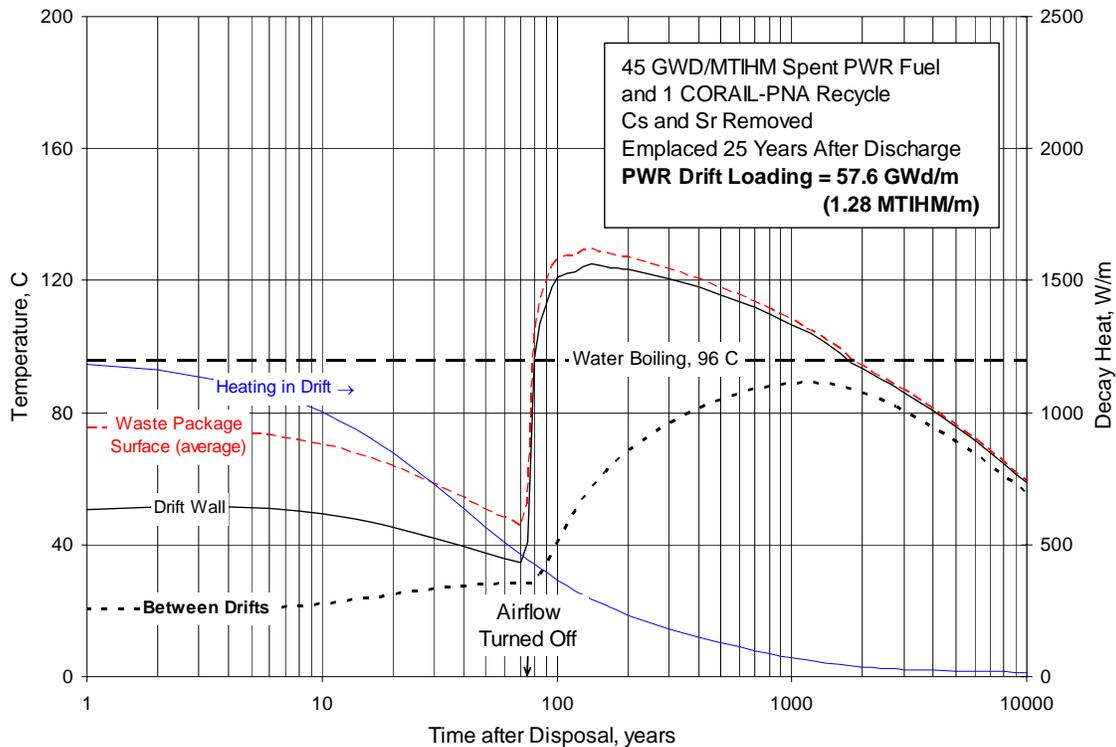


Figure 14. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the CORAIL-PNA-1 group, 45 GWd/MTIHM discharge burnup.

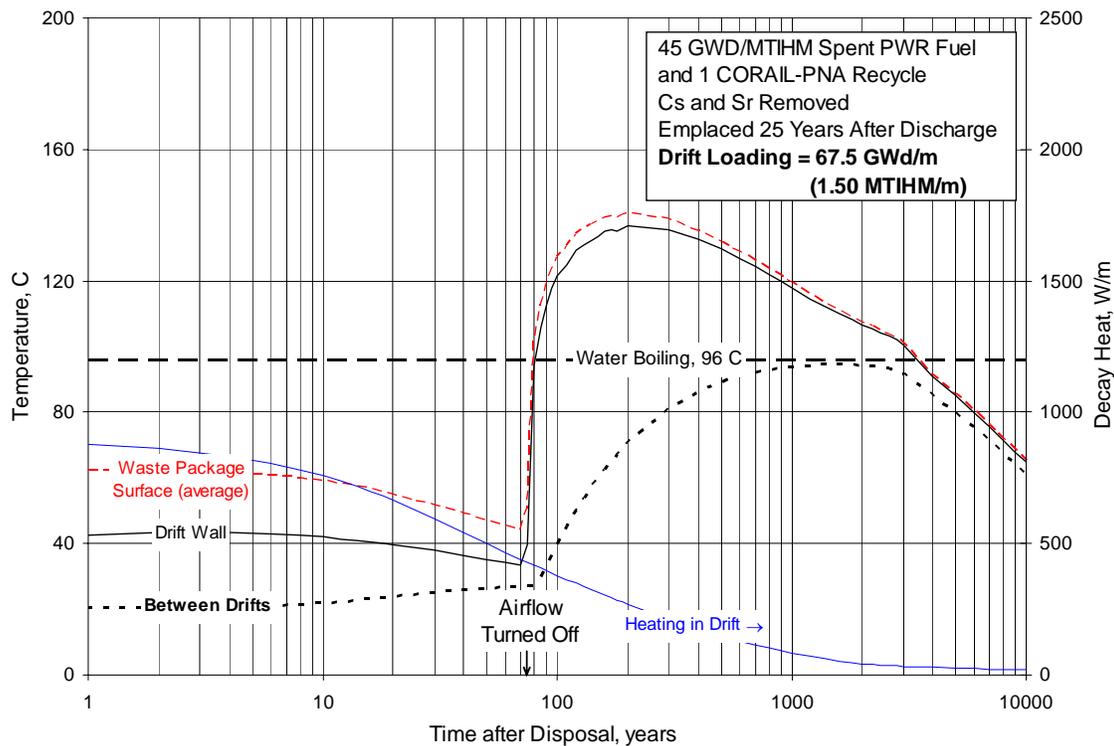


Figure 15. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the CORAIL-PNA-1 group, 45 GWd/MTIHM discharge burnup; loading increased to a maximum drift loading of 1.50 MTIHM/m

To understand the nature of the difference in benefit between the first MOX recycle and the first CORAIL-PNA recycle, it is useful to examine the details of the decay heat characteristics for the CORAIL-PNA-1 group as shown in Figure 16, where a number of small, but significant, differences can be observed. First, there is a greater reduction in the Am-241 content between the spent PWR fuel group and CORAIL-PNA-1 (C-P-1) group, indicating that relatively less Pu-241 is present in the discharged fuel. Also, there is a smaller increase in the Pu-238 content relative to the reference case of spent PWR fuel. There is also a greater reduction in decay heat from Pu-239 and Pu-240 content, as well as Cm-242. These changes in the isotopic composition allow a faster decrease in the decay heat after 100 years, when compared to either the MOX-1 case or the spent PWR fuel case, and results in the larger allowable increase in the drift loading. It would seem at this time that the design of a hybrid assembly, using uranium of higher enrichment to provide much of the fissile content rather than relying entirely on the recycled plutonium, americium, and neptunium as in MOX recycling, allows a more favorable shift in the discharge isotopic composition to benefit the thermal load in the repository due to the lower plutonium enrichment.

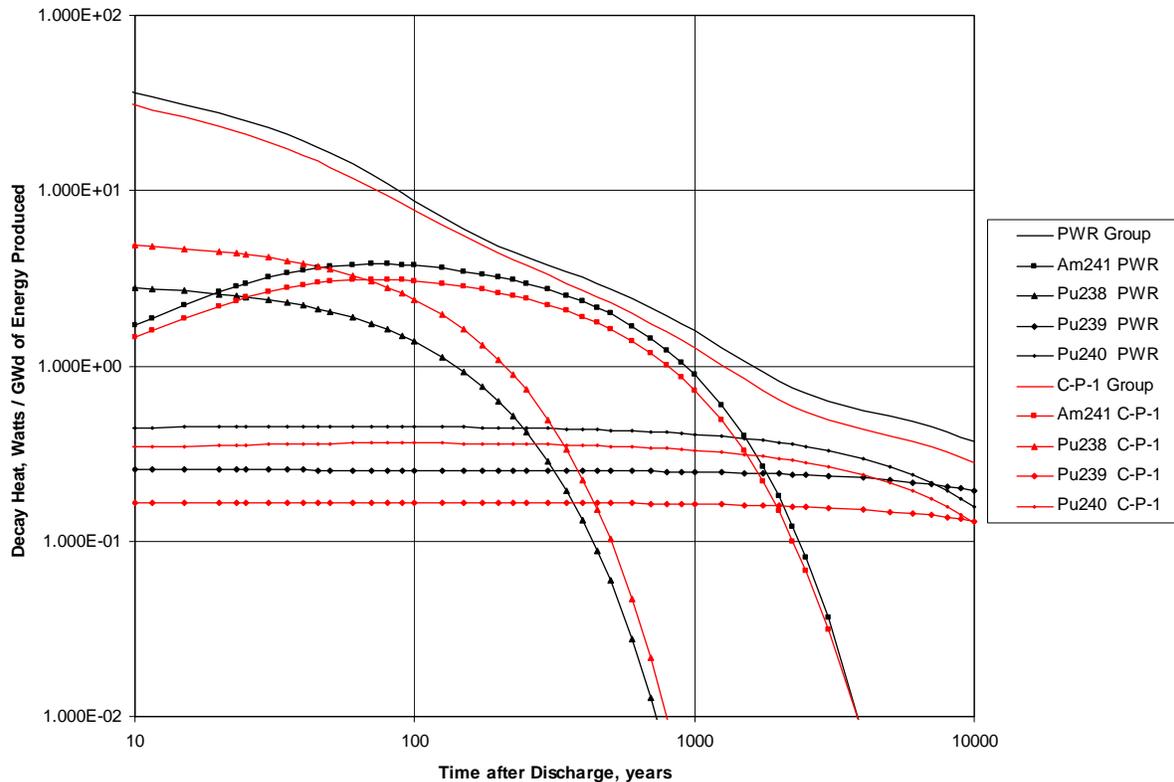


Figure 16. Decay heat characteristics for the CORAIL-PNA-1 group compared to the spent PWR fuel group, assemblies irradiated to 45 GWd/MTIHM discharge burnup

III.C CORAIL-PNA-2

CORAIL-PNA-2 is created by processing the entire CORAIL-PNA-1 assembly at the same time, mixing the irradiated enriched uranium pins with the irradiated recycled plutonium, americium, and neptunium pins. The recovered plutonium, americium, and neptunium from a single CORAIL-PNA-1 assembly is used to fabricate the 32% of the pins containing the recycled material for a single CORAIL-PNA-2 assembly, with the remaining 68% of the pins being fabricated from enriched uranium, having the same increased enrichment of 4.87% as for the CORAIL-PNA-1 case. The CORAIL-PNA-2 group is created by combining the processing waste from both a spent PWR fuel assembly and a spent CORAIL-PNA-1 fuel assembly, minus the cesium and strontium, plus the direct disposal of the irradiated CORAIL-PNA-2 assembly. The behavior of the repository in response to this group is compared to the response for a comparable group of 3 spent PWR fuel assemblies to preserve the total integrated energy.

The transient temperature behavior of the repository for the CORAIL-PNA-2 group is shown in Figure 17. As with CORAIL-PNA-1, the overall transient behavior is similar to the reference case, with loading still limited by the peak temperature midway between adjacent drifts. However, the maximum drift loading can be further increased to 1.70 MTIHM/m (representing material that produced 76.5 GWd of energy being stored per meter of drift). Examining the decay heat characteristics shows that the trends observed between CORAIL-PNA-1 and spent PWR fuels are continued, as shown in Fig. 18, with larger decreases in Am-241, Pu-240, and Pu-239 and only a small increase in Pu-238.

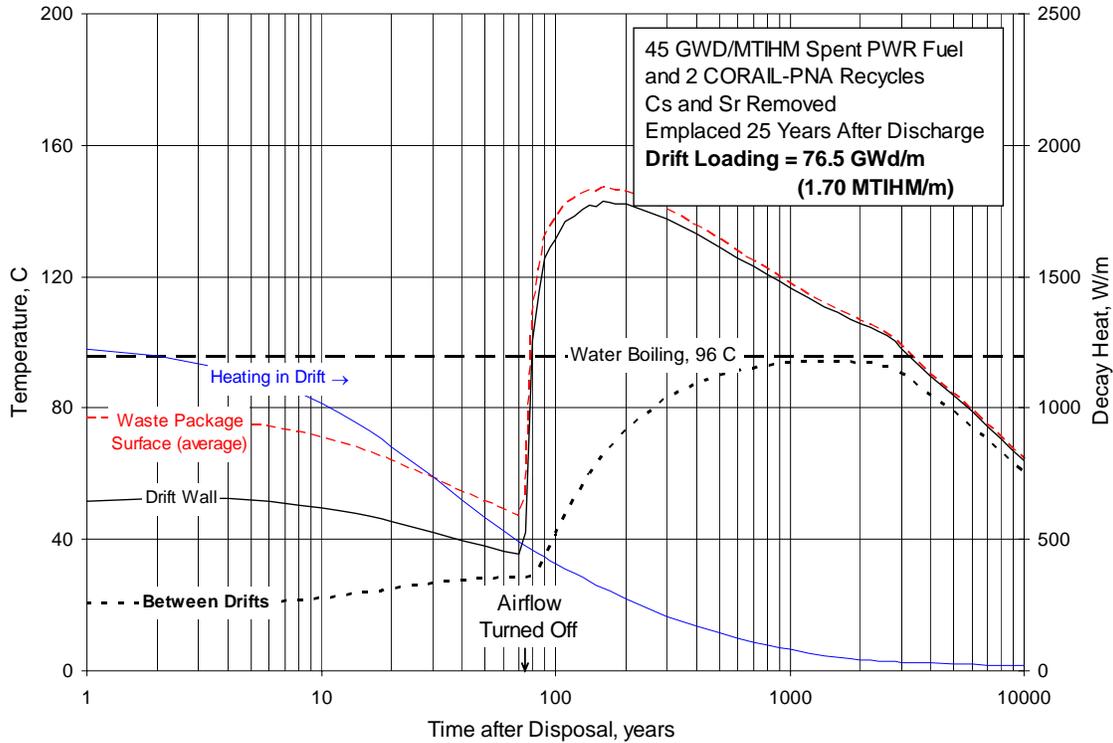


Figure 17. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the CORAIL-PNA-2 group, 45 GWd/MTIHM discharge burnup; loading increased to a maximum drift loading of 1.70 MTIHM/m.

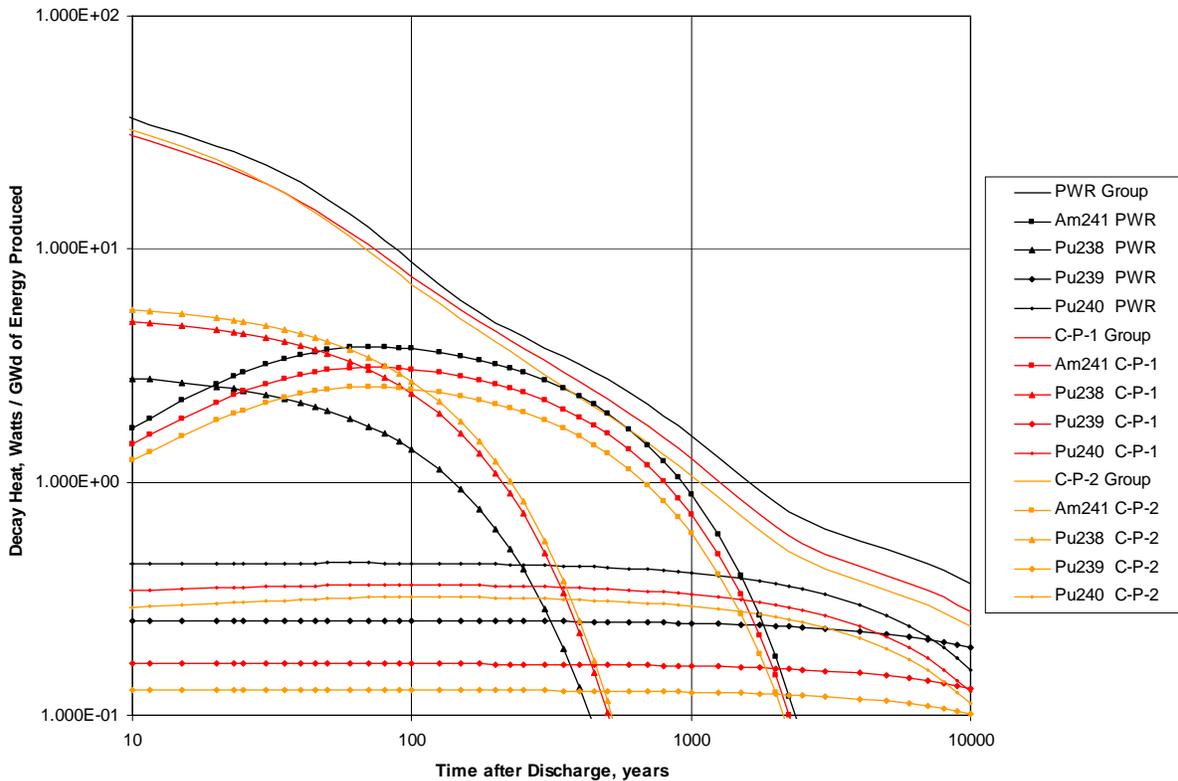


Figure 18. Decay heat characteristics for the CORAIL-PNA-2 group compared to the spent PWR fuel group, assemblies irradiated to 45 GWd/MTIHM discharge burnup

IV.D CORAIL-PNA-3

The third CORAIL-PNA pass is fabricated from processing CORAIL-PNA-2 assemblies and using the recovered plutonium, americium, and neptunium to fabricate new pins, each CORAIL-2 assembly providing about 32% of the pins for a single CORAIL-PNA-3 assembly. The remaining pins are fabricated from enriched uranium, but increased again to 4.92% in this case. The CORAIL-PNA-3 group is composed of the irradiated CORAIL-PNA-3 assembly along with the waste from processing a CORAIL-PNA-2, a CORAIL-PNA-1, and a spent PWR fuel assembly, with the cesium and strontium removed from the process waste at each step. The repository response to this group is compared with the repository behavior for a group of 4 spent PWR fuel assemblies, preserving the total integrated energy between the two groups.

As with the previous CORAIL-PNA cases, the overall transient behavior of the repository is the same, with the repository loading being limited by the peak temperature midway between adjacent drifts, although the lower decay heat results in a lower peak temperature. Therefore, the maximum allowable loading has been further increased to 1.90 MTIHM/m, representing material that produced 85.5 GWd of energy being stored per meter of drift. As mentioned in the previous paragraph, the lower decay heat is caused by the continuing trend of lower Am-241, Pu-240, and Pu-239, with little increase in the Pu-238 contribution.

IV.E CORAIL-PNA-4

The CORAIL-PNA-4 case is the next recycle using the CORAIL concept, where the recovered plutonium, americium, and neptunium from each CORAIL-PNA-3 assembly is used to fabricate 32% of the pins in each CORAIL-PNA-4 assembly. The remaining pins are fabricated from enriched uranium, increased to 4.95%. The CORAIL-PNA-4 group is composed of the irradiated CORAIL-PNA-4 assembly, and the process waste from processing a CORAIL-PNA-3, a CORAIL-PNA-2, a CORAIL-PNA-1, and a spent PWR fuel assembly. Cesium and strontium are removed from all process waste for separate storage. The repository behavior for this group is compared with that for a group of 5 spent PWR fuel assemblies, preserving the total integrated energy between the two groups.

As with the other CORAIL-PNA cases, the overall transient behavior of the repository is the same, with the repository loading being limited by the peak temperature midway between adjacent drifts, although the decay heat is lower than the previous CORAIL-PNA cases. As a result, the maximum allowable drift loading has been increased to 2.05 MTIHM/m, representing material that produced 92.3 GWd of energy being stored per meter of drift.

IV.F CORAIL-PNA-5

The CORAIL-PNA-5 case is constructed in the same manner as the previous cases, where this next recycling pass using the CORAIL concept takes the recovered plutonium, americium, and neptunium from the CORAIL-PNA-4 assembly to make some of the pins in the CORAIL-PNA-5 assembly. The uranium enrichment needs in the other pins continues to increase slowly, now being up to 4.98%. The resulting decay heat for the CORAIL-PNA-5 group is reduced further from the previous case, with the maximum allowable loading increased to 2.20 MTIHM/m, representing material that produced 99 GWd of energy being stored per meter of drift.

IV.G CORAIL-PNA-6

The CORAIL-PNA-6 case is constructed as the previous cases, where this next recycling pass using the CORAIL concept takes the recovered plutonium, americium, and neptunium from the CORAIL-PNA-5 assembly to make some of the pins in the CORAIL-PNA-6 assembly. The uranium enrichment needs in the other pins continues to increase slowly, now being up to 5.00%. The resulting decay heat for the CORAIL-PNA-6 group is again reduced from the previous case. The maximum allowable loading has now been increased to 2.30 MTIHM/m, representing material that produced 103.5 GWd of energy being stored per meter of drift.

IV.H CORAIL-PNA-7

The seventh pass of CORAIL-PNA was the last CORAIL recycle examined as part of this study. The CORAIL-PNA-7 case is constructed the same as the previous cases where this next recycling pass using the CORAIL concept takes the recovered plutonium, americium, and neptunium from the CORAIL-PNA-6 assembly to make some of the pins in the CORAIL-PNA-7 assembly. The uranium enrichment needs in the other pins continues to increase slowly, now being up to 5.02%. The CORAIL-PNA-7 group is composed of the irradiated CORAIL-PNA-7 assembly, and the process waste from processing a CORAIL-PNA-6, CORAIL-PNA-5, CORAIL-PNA-4, CORAIL-PNA-3, CORAIL-PNA-2, CORAIL-PNA-1, and a spent PWR fuel assembly. Cesium and strontium are removed all process waste for separate storage. The resulting decay heat for the CORAIL-PNA-7 group is again reduced from the previous case.

The transient temperature behavior of the repository for the CORAIL-PNA-7 group is given in Fig. 19. The drift loading has been maintained at the reference value for spent PWR fuel of 1.20 MTIHM/m. While the overall transient behavior with the CORAIL-PNA-7 group is similar, the peak temperature is much lower at about 62 °C. This allows the drift loading to be substantially increased up to 2.40 MTIHM/m, (representing material that produced 108 GWd of energy being stored per meter of drift). The results for the increased maximum loading are shown in Fig. 20, where it is observed that the overall transient behavior is very similar to the reference case.

IV.G Summary of CORAIL-PNA Results

The benefit using the CORAIL-PNA approach is summarized in Figure 21. There is a faster increase in drift loading with each CORAIL-PNA recycle than was obtained with the MOX recycling approach. The hybrid scheme of using both enriched uranium fuel pins and fuel pins that contain recycled higher actinide elements in each assembly seems to be able to enhance the power production of the recycle material while achieving a favorable balance of the more troublesome isotopes compared to once-through PWR fuel. In addition, unlike the MOX recycling cases, the repeated recycling using the CORAIL-PNA concept do not appear to pose a problem with the reactor safety coefficients or fuel enrichment, and may represent a more practical way of using the recovered plutonium, americium, and neptunium. It should also be noted, however, that the temperature peak immediately after closure of the repository is getting higher with each recycle, as happened with MOX. This implies that with enough recycling, the repository may become limited by this temperature constraint as well, and the ability to further increase repository loading may be hampered unless additional separations are performed.

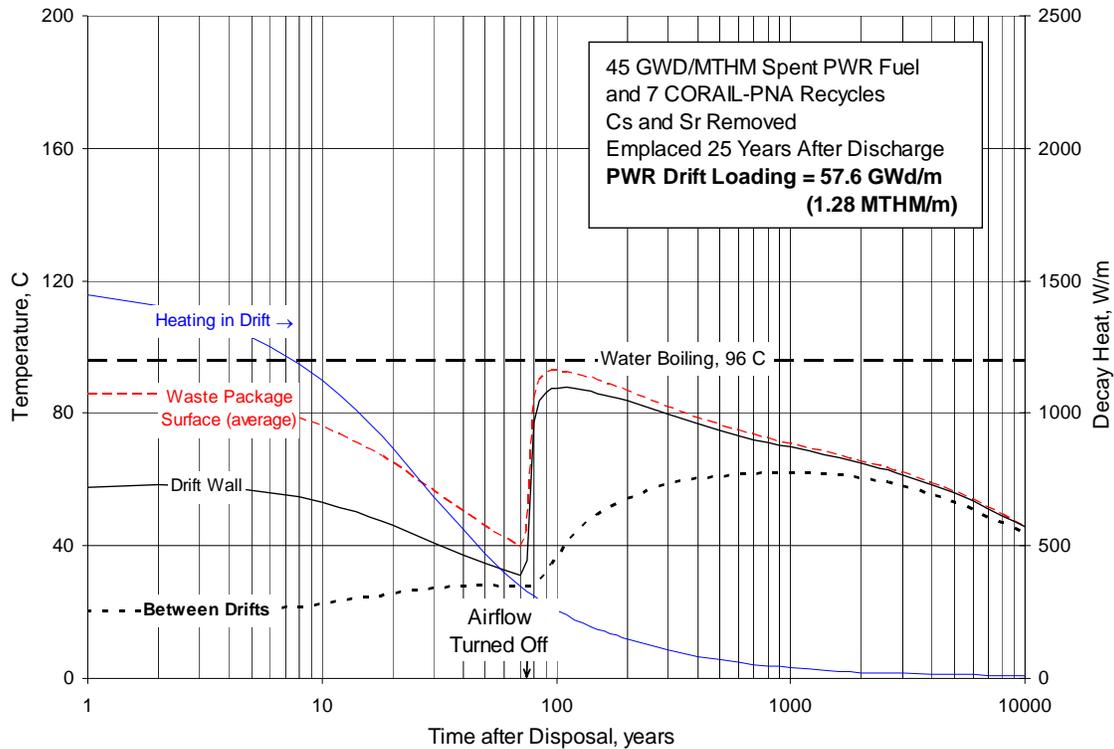


Figure 19. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the CORAIL-PNA-7 group, 45 GWd/MTIHM discharge burnup.

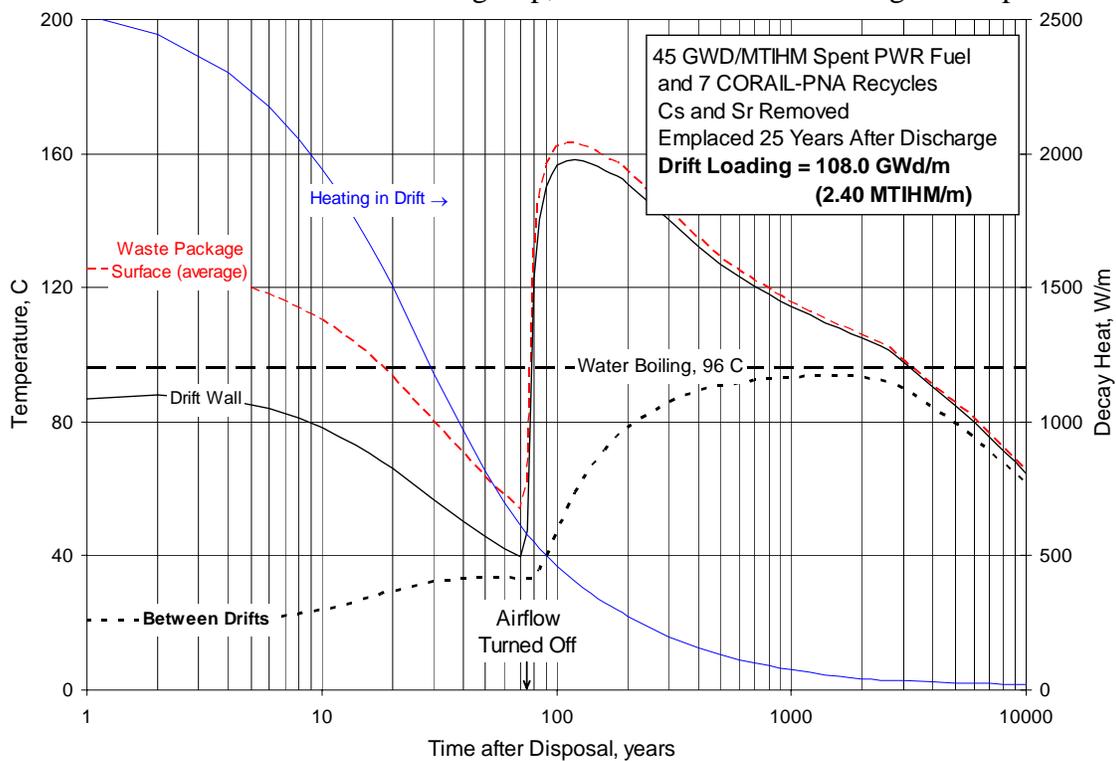


Figure 20. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the CORAIL-PNA-7 group, 45 GWd/MTIHM discharge burnup; loading increased to a maximum drift loading of 2.40 MTIHM/m.

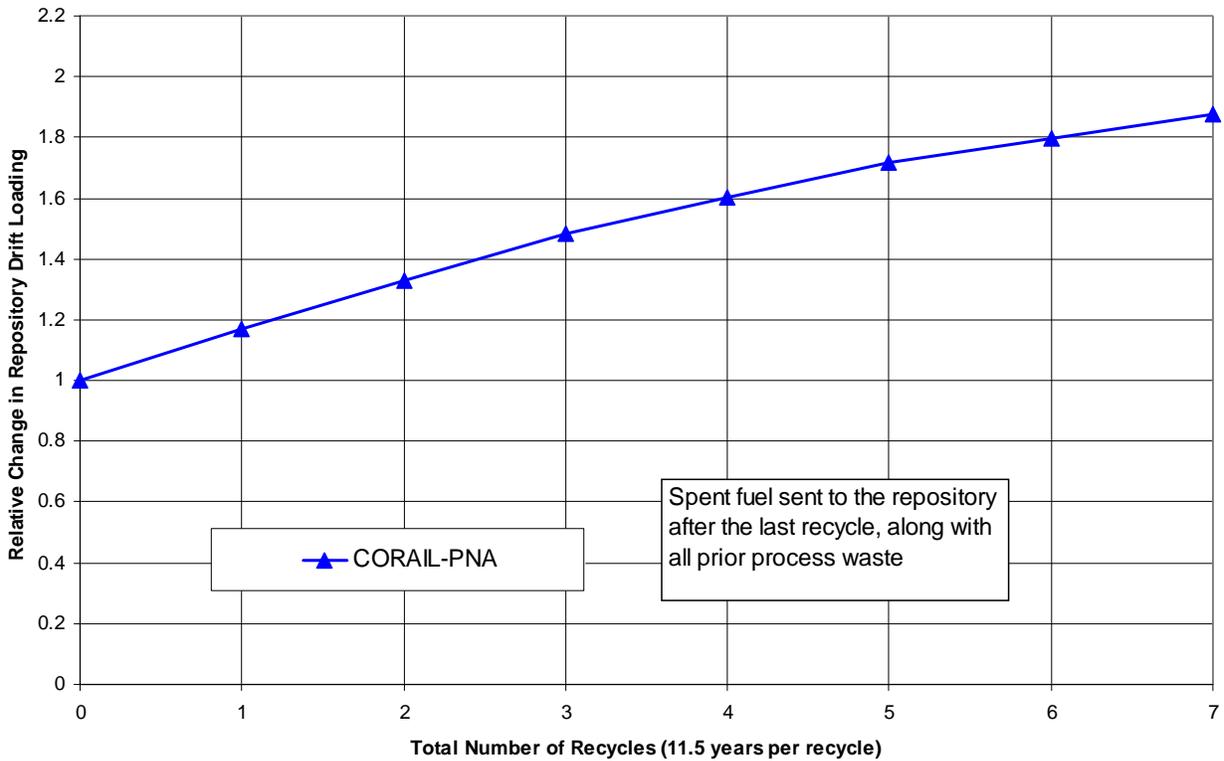


Figure 21. Impact of using CORAIL recycling on the linear loading of a repository at Yucca Mountain.

V. Recycling of Plutonium, Americium, and Neptunium as Inert Matrix Fuel (IMF)

Another approach to the recycling of plutonium, americium, and neptunium is to do something similar to the MOX recycling, but to use an inert fuel matrix instead of uranium oxide. For the purposes of these analyses, zirconia has been used as the fuel matrix. While this fuel may be referred to casually as ‘non-fertile’ fuel, actually the fuel makes more use of the fertile isotopes of plutonium than the other recycling strategies. The fertile isotopes are a very important source of fissionable material as the irradiation cycle proceeds. However, due to the inert matrix, the amount of heavy metal in each fuel assembly is much less than for a standard PWR assembly due to the absence of uranium. In order to provide the same integrated energy from such an assembly, the burnup for IMF fuel would therefore be much higher on a mass basis. For example, in the first pass of IMF fuel, there is only 42.3 kg of heavy metal in each assembly, compared with 461.3 kg in a standard PWR assembly. The burnup achieved with the first pass of IMF fuel is 551 GWd/MTIHM, as compared to 51 GWd/MTIHM for the PWR and MOX fuel, but integrated assembly energy is the same.

V.A IMF-1

As stated above, the IMF-1 fuel is composed entirely of plutonium, americium, and neptunium in a matrix of zirconia. To recover sufficient fissile material so that the IMF assembly can achieve the same integrated energy as a PWR assembly, 7.11 assemblies of spent PWR fuel are needed. As stated above, the IMF assembly only contains 42.3 kg of heavy metal, compared to the 461.3

kg of heavy metal in a PWR assembly. It is important to note that this is about half of the plutonium, americium, and neptunium loading used in the MOX case, and it appears that this is the result of not having the fertile U-238 in the fuel, so that less fissile material is needed for the reactor to go critical. The lower loading of plutonium, americium, and neptunium is important to the ability of the IMF fuel to benefit the repository. The IMF-1 group is composed of the spent IMF-1 fuel assembly, along with the waste from processing 7.11 spent PWR fuel assemblies. The cesium and strontium is removed from the process waste. As with the other cases, curium remains with the process waste and is not recycled.

The transient temperature performance of the IMF-1 group in the repository at the reference loading of 58.6 GWd/m (equivalent to 1.15 MTIHM/m for the direct disposal of spent PWR fuel) is shown in Figure 22. The peak temperature is much lower than for the direct disposal of spent PWR fuel at 65 °C, indicating that a substantial increase in drift loading is possible, up to a loading that represents material that produced about 105 GWd of energy being stored per meter of drift, an increase of 1.8 times the value obtained in the spent PWR fuel reference case. The results for this case are shown in Figure 23.

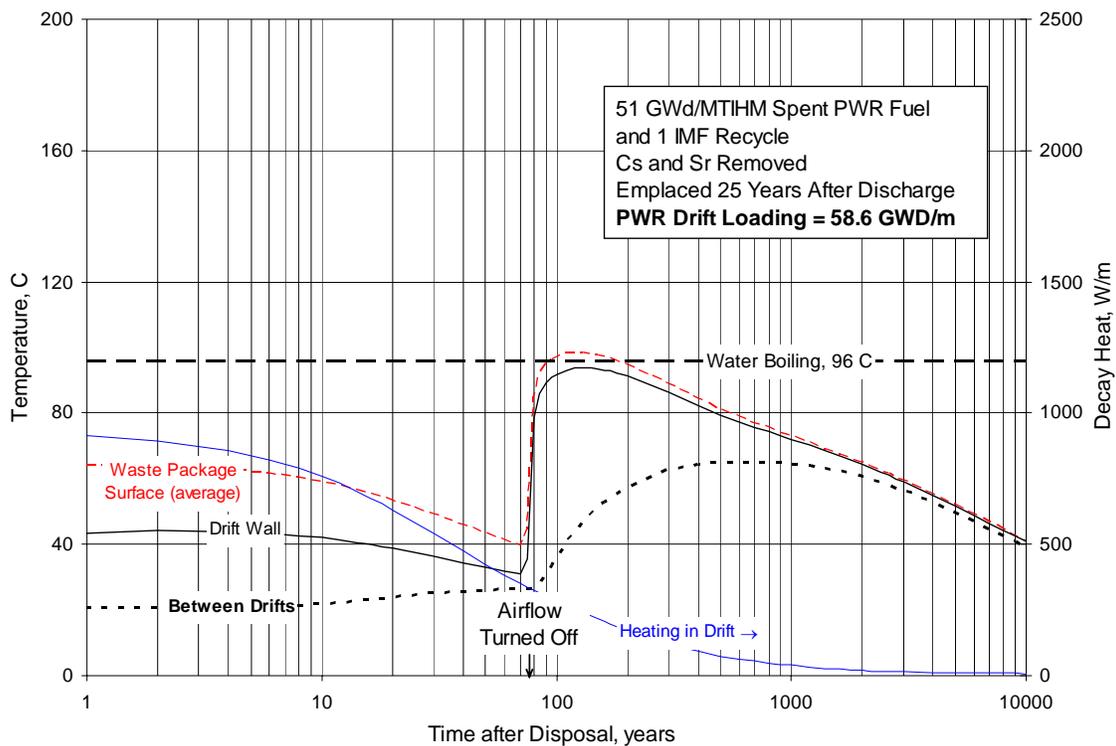


Figure 22. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the IMF-1 group at the reference drift loading of 58.6 GWd/m

When the assemblies are grouped, with 1 IMF-1 spent fuel assembly and the processing waste from 7.11 spent PWR fuel assemblies, the amount of heat from the relevant plutonium isotopes is much smaller for the IMF-1 case than for the MOX-1 case, per unit energy, as shown in Figure 24. There is a large reduction in the amount of decay heat from Am-241, reflecting the much lower Pu-241 content at discharge. This is partially offset by the increase in Pu-238, similar to that seen for the MOX fuel cases. There are also sizeable reductions in the amount of Pu-239 and Pu-240 for IMF-1. In total the decay heat from the heavy metal elements is about half of

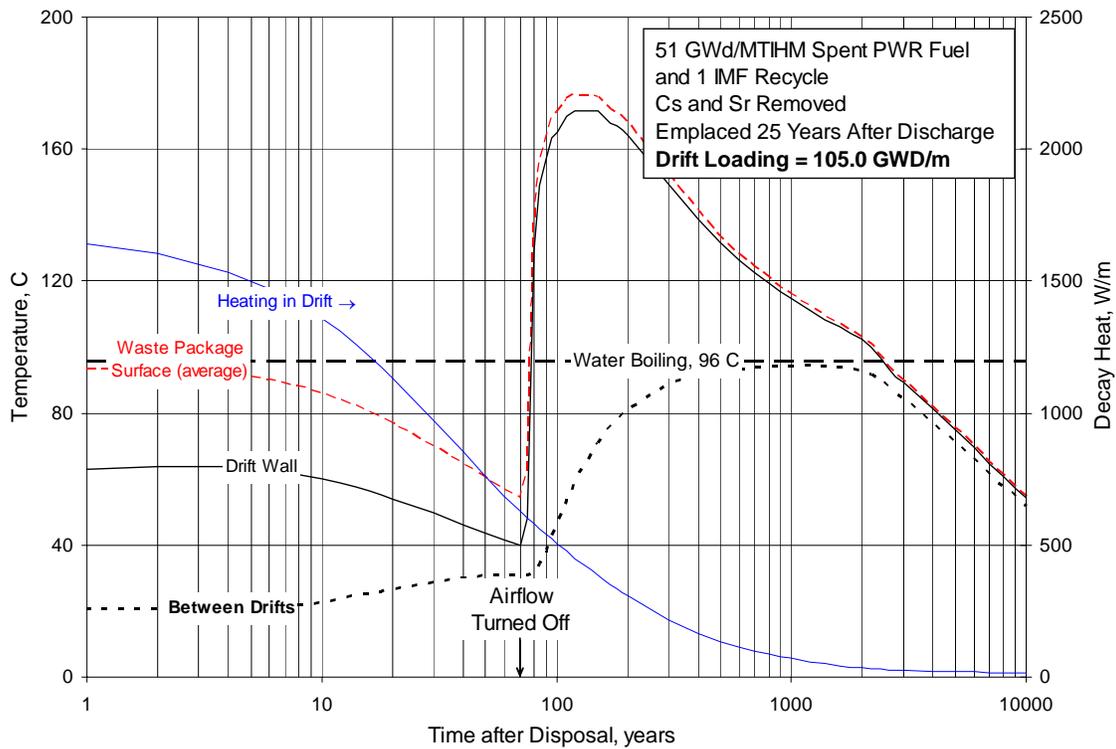


Figure 23. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the IMF-1 group; increased to maximum drift loading of 105.0 GWd/m

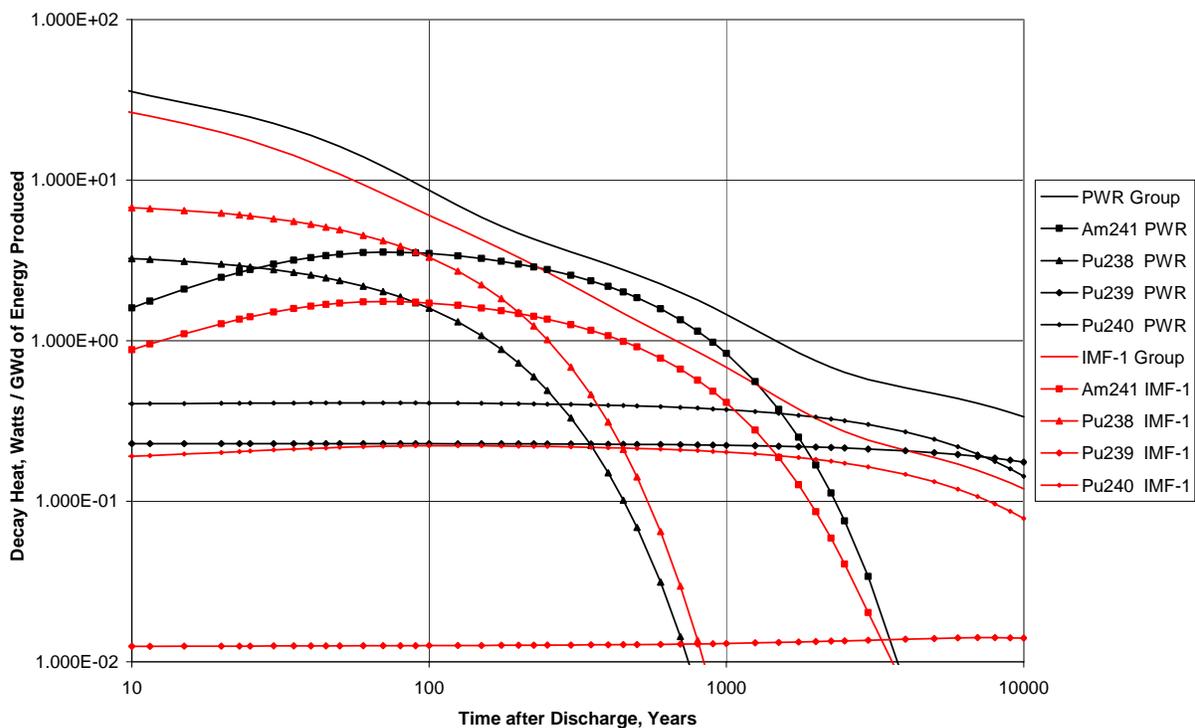


Figure 24. Decay heat characteristics for the IMF-1 group compared to the spent PWR fuel group

that obtained for the direct disposal of spent PWR fuel, and is the reason that drift loading can be increased by a factor of 1.8 after only the first IMF irradiation.

This is a much better performance in a single pass than either the MOX or CORAIL strategies. However, as with some of the other cases, at this stage of the investigation, the best practical method of using IMF assemblies still needs to be evaluated, whether in a homogeneous or heterogeneous reactor core, or whether to even use heterogeneous assemblies as with the CORAIL approach, but with some IMF fuel pins.

V.B IMF-2

For the second recycle, the amount of plutonium, americium, and neptunium that goes into each assembly is far higher than for IMF-1, 348 kg vs. 42.3 kg, due to high depletion of Pu-239 during the IMF-1 pass. The remaining matrix is again zirconia. The fabrication of one IMF-2 assembly requires the processing of 19.9 IMF-1 assemblies to obtain sufficient fissile material. The IMF-2 group consists of the irradiated IMF-2 assembly, along with the processing waste from 19.9 IMF-1 assemblies and 141 spent PWR assemblies. As before, the cesium and strontium are removed from the process waste. The transient performance of the repository with the IMF-2 group is shown in Figure 25. As the figure shows, the peak temperature midway between adjacent drifts has been reduced to 59 °C due to the lower decay heat for the IMF-2 group.

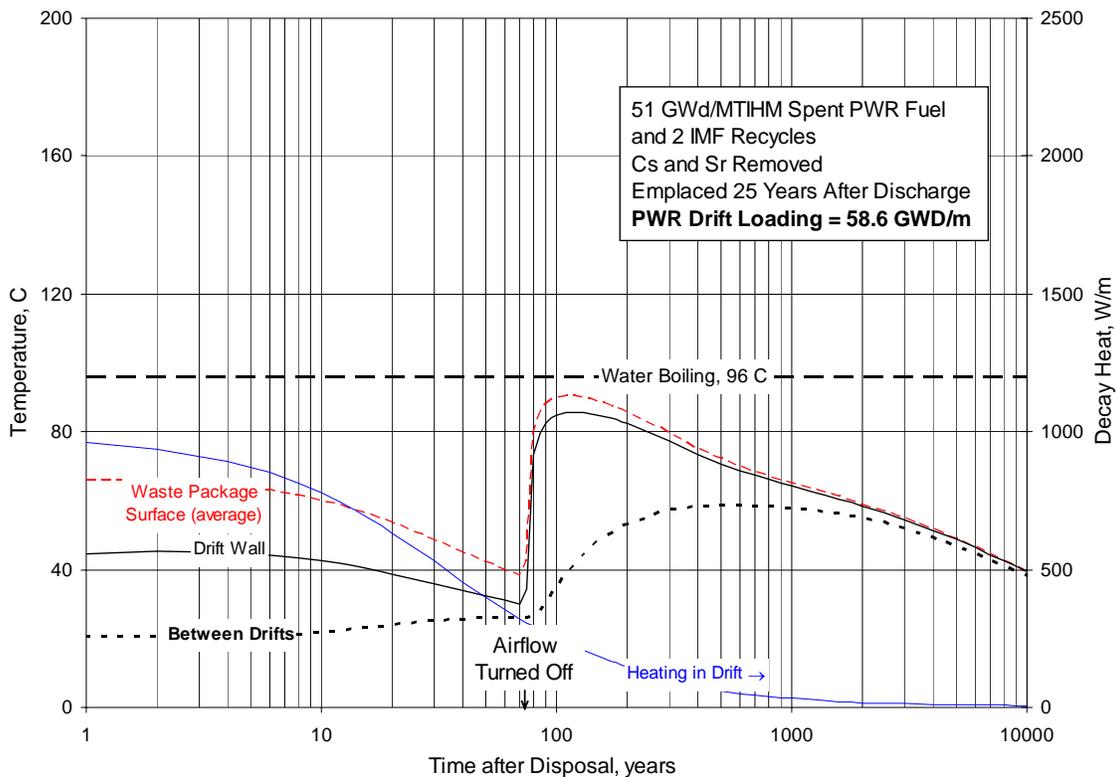


Figure 25. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the IMF-2 group; reference loading of 58.6 GWd/m.

The lower decay heat for the IMF-2 group allows an increase in maximum drift loading with material representing 125 GWd of energy produced being stored per meter of drift. This is an increase of about 2.1 compared to the spent PWR fuel. It is important to note that the incremental benefit from the second IMF pass is much less than it was for the first IMF pass. The transient repository response for this loading is shown on Fig. 26. A third IMF pass was attempted, but even with pins fabricated entirely from the recycled plutonium, americium, and neptunium from the IMF-2 assemblies, with no zirconia matrix at all, there was not enough fissile material left to achieve the same integrated energy for the assembly.[4] At the time of discharge, the reduction in decay heat per GWd of energy produced was only slightly smaller than that for IMF-2, indicating that there would be very little further benefit from recycling material using inert matrix fuel beyond the second pass.

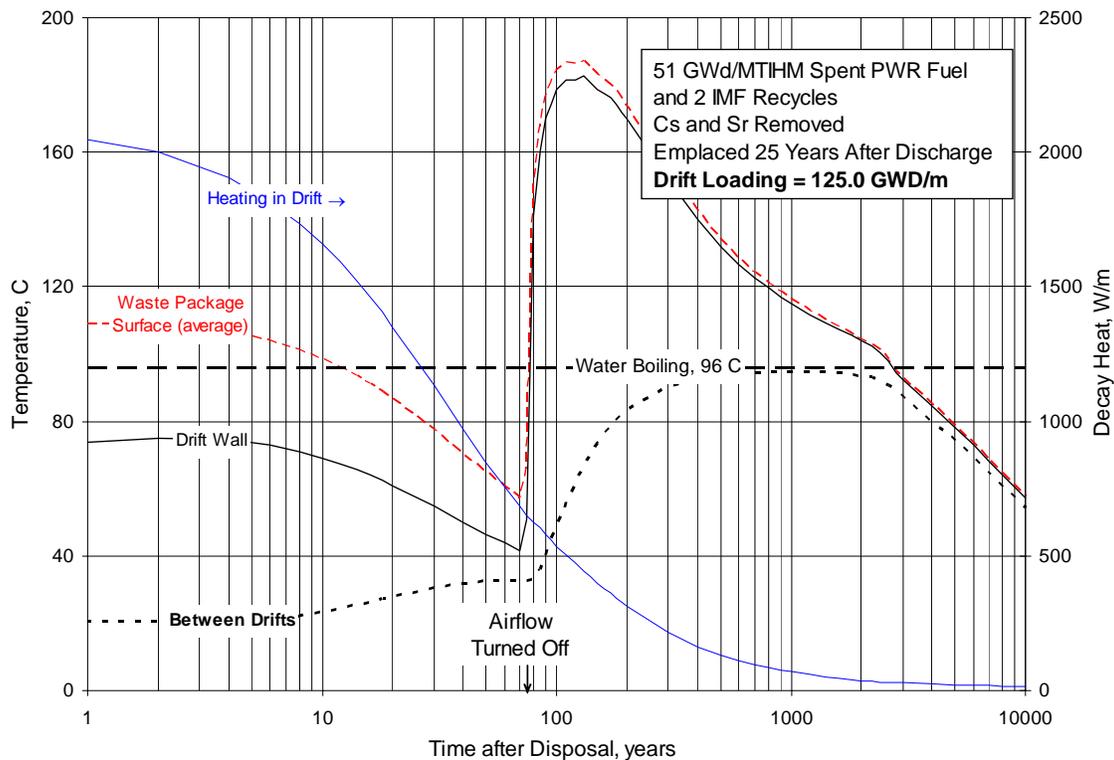


Figure 26. Transient temperature response of a drift in the central region of a repository at Yucca Mountain for the IMF-2 group; loading increased to a maximum drift loading of 125.0 GWd/m.

It is also useful to note on Figure 26 that the temperature peak at about 125 years is approaching the 200 °C temperature limit for the drift wall temperature, suggesting that further benefit from additional recycles may be limited for this reason as well.

Whether the difficulties observed can be overcome by using heterogeneous assemblies or reactor cores is unknown at this time. It may be that it would be necessary to blend in new fissile material with the discharged IMF-2 material to be able to continue the irradiations.

VI. Sensitivity of Repository Drift Loading Increase to the Age of the Spent Fuel

Since the repository drift loading for direct disposal of spent PWR fuel is controlled by the decay heat from americium (Am-241) arising from the decay of Pu-241 in the stored spent fuel after irradiation, the analyses reported in the previous section were performed for processing all spent fuel at 5 years after discharge to minimize the decay of Pu-241 and thus maximize the benefit to the repository. The benefit to the repository is also closely related to the total amount of Pu-241 (and Am-241) in the discharged fuel when the spent fuel is directly disposed in the repository. Higher enrichments are needed when older fuel is processed and recycled due to the decay of fissionable isotopes and the increase in isotopes that are more likely to capture neutrons in the thermal spectrum. The higher enrichment increases both the Pu-239 and Pu-240 content in the recycled fuel, and the net production of Pu-241 is increased, potentially reducing the benefit to the repository from processing and recycling spent fuel. To quantify the sensitivity of the repository drift loading increase to the age of the spent fuel, analyses were also conducted for one recycle in each case using spent fuel that had been stored for 20 years. (This part of the study was performed subsequent to the initial report to the ANTT Subcommittee of NERAC, and was related to ANTT Subcommittee questions on the effects of extended cooling prior to processing. The effort in this area was covered by a separate AFCI work package, but is included in this report for completeness of the descriptions of the repository benefits evaluations.) The results are summarized in Figure 2.

Table 2. Repository drift loading increase factors for one recycle using MOX, CORAIL-PNA, and IMF with spent PWR fuel processed at 5 years and at 20 years.

Recycle Strategy	5 Year Old Spent PWR Fuel	20 Year Old Spent PWR Fuel
MOX	1.09	1.04
CORAIL-PNA	1.22	1.16
IMF	1.79	1.19

As is shown in the table, the effect of processing older fuel and recycling the recovered plutonium, americium, and neptunium in an LWR differs depending on the recycling strategy. The least impact is observed for CORAIL-PNA, where the use of enriched uranium reduces the sensitivity to the changing isotopic composition of the recycled materials by allowing lower plutonium enrichment. The reduction in benefit as measured by the increase in repository drift loading is slightly less than 30%. The greatest impact is observed with IMF, since IMF depends entirely on the fissile content of the recovered materials, which is mostly plutonium, and more fissile is needed as the composition and isotopic distribution changes with age. In this case, the reduction in drift loading benefit is over 75%. The effect on MOX is between these two, with a reduction in the allowable drift loading increase of about 50%, due to the use of a uranium matrix that provides additional fissile material during irradiation. However, the overall effect is that for a single recycle step using older fuel, the use of IMF is comparable to using CORAIL-PNA, with the benefit of using MOX becoming minimal.

VII. Repository Benefits from Process Waste and Repository Limitations from Spent Fuel

In order to understand the source of the limitation arising from the limited recycle strategies, it is useful to consider the behavior of the repository for each component of a recycling group as studied in this report. Using the MOX-1 group as an example, the potential repository benefit has been determined for each component of the MOX-1 group, namely the process waste from 13.5 spent PWR fuel assemblies as one part, and the irradiated MOX-1 assembly as the other part. The repository response to the process waste from the spent PWR fuel is shown in Fig. 27. As is shown, the allowable increase in drift loading is substantial, with a maximum drift loading of the waste from processing 59.1 MTIHM being stored per meter of drift, representing material that produced 3010 GWd of energy being stored per meter of drift. This is an increase of a factor of 51 over the direct disposal of spent PWR fuel, and consistent with the expectations described in Section I.A. The temperature limit preventing the loading from being even higher now occurs immediately at the time of placement, where the drift wall temperature is just below 200 °C. The source of the heat causing this high temperature is curium in the process waste, and removing the curium would provide additional benefit.

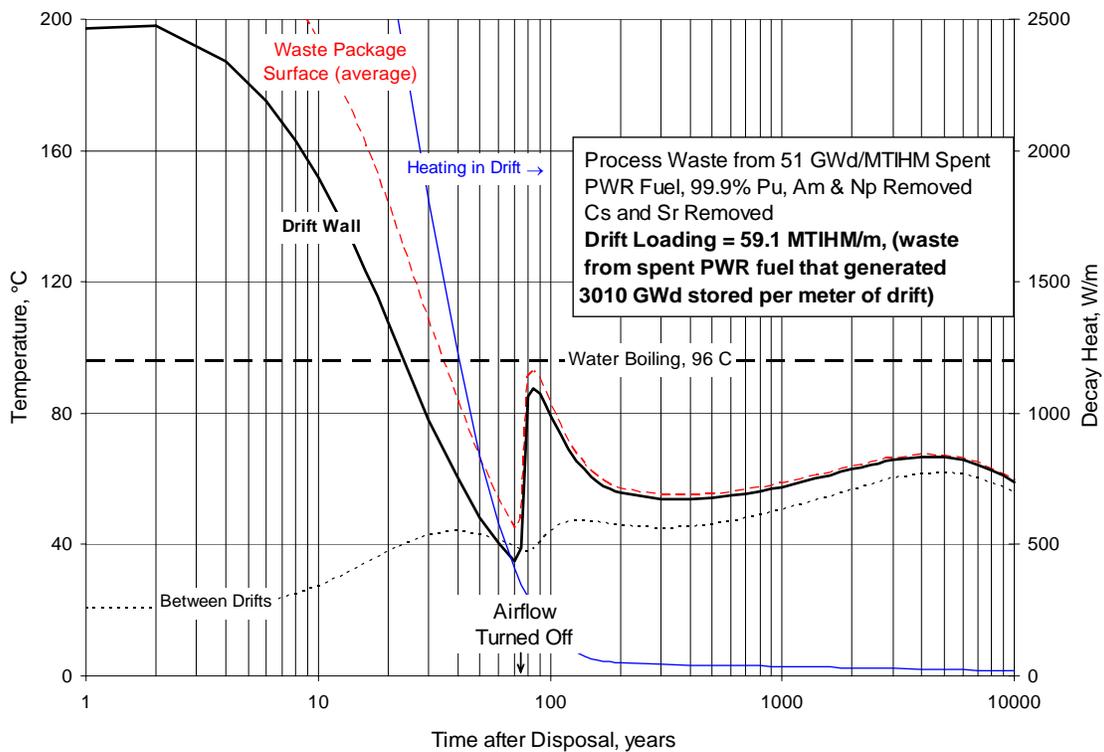


Figure 27. Transient temperature response for a drift in the central region of a repository at Yucca Mountain for the process waste from spent PWR fuel, 51 GWd/MTIHM; loading increased maximum loading of process waste from 59.1 MTIHM being stored per meter.

However, in studying the repository response associated with the disposal of the one spent MOX-1 assembly, a far different result is obtained. The transient repository response for this case is shown in Figure 28. Due to the high long-term decay heat of this assembly after irradiation, it is only possible to load this material to about 0.08 MTIHM per meter of drift,

representing material that produced 4.33 GWd of energy being stored per meter of drift, while remaining below the 96 °C temperature limit midway between adjacent drifts. This is a reduction in drift loading of a factor of 13.5 as compared to the direct disposal of spent PWR fuel. When the spent MOX-1 assembly and the process waste from the 13.5 spent PWR fuel assemblies are taken together, the overall benefit for the MOX-1 group is only a factor of 1.087.

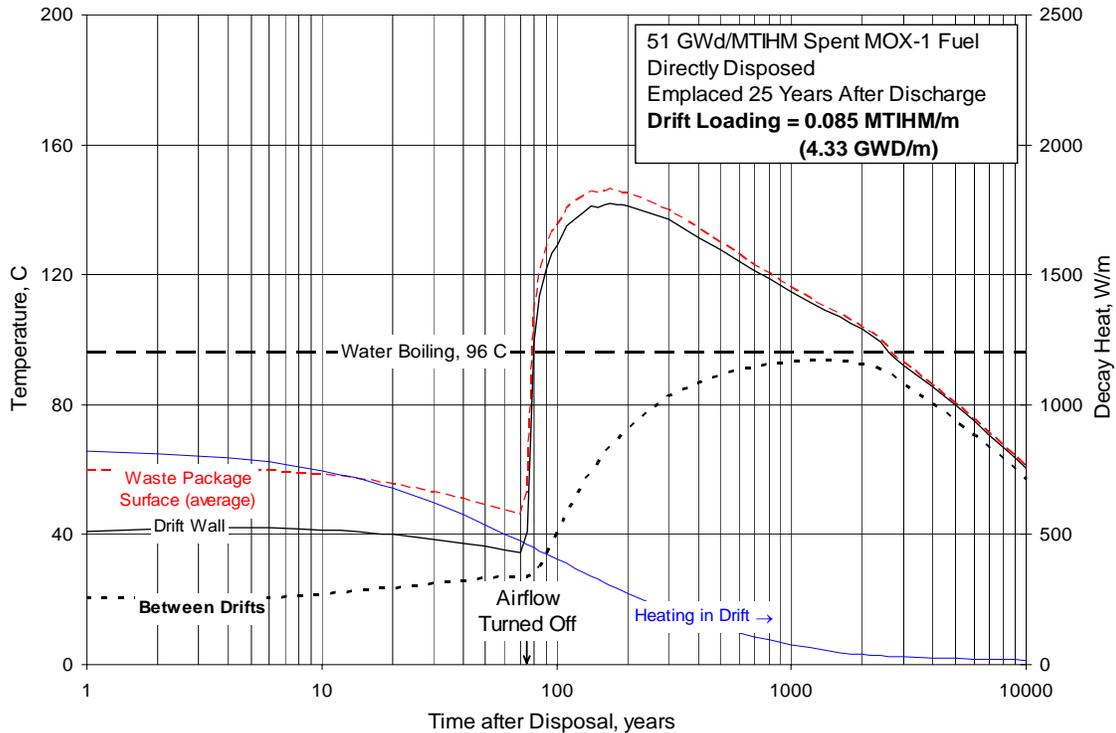


Figure 28. Transient temperature response for a drift in the central region of a repository at Yucca Mountain for spent MOX-1 assemblies, 51 GWd/MTIHM; decreased loading of 0.085 MTIHM/m.

These results highlight that **it is the need to directly dispose of the last assemblies (with all of the actinide elements that they contain) for strategies with a limited number of recycles that truly limits the benefits to increasing the repository capacity.** This effect can also be seen by examining the results presented in Table 3, where the charge and discharge plutonium composition is presented for each type of assembly, on a MTIHM basis (or the equivalent for the IMF-1 assembly). First, it is noted that in all cases, the discharge material after the first pass is either higher or significantly higher than the amount in spent PWR fuel. This is due to the concentration of actinide elements that occurs in fabricating the MOX, CORAIL-PNA, or IMF assemblies. It would be expected that each of the assemblies with such discharge compositions would require significantly more space in the repository than a spent PWR fuel assembly. This detrimental effect is offset in all cases by the large gains in loading for the assemblies that have been processed, where only the process waste from those assemblies is sent to the repository.

Similar analyses can be conducted for Am-241, which show that the use of IMF fuel transmutes about half of the Am-241 during irradiation, while MOX fuel actually gains in Am-241 content

during irradiation. These factors all contribute to the ability of each strategy to provide a benefit to the repository.

Considering the details on plutonium for each case, the extreme burnup of the Pu-239 with the IMF-1 fuel is evident, as are the relative increases and decreases of the plutonium isotopes for the other fuel types. The consumption of the Pu-240 fertile isotope in the IMF-1 case is also shown, although there are gains in both the Pu-238 and Pu-242. It should also be mentioned that the recycling of the neptunium, which does not impact the repository loading directly since it has very little decay heat, had some unanticipated consequences. First, the presence of the neptunium required a higher plutonium loading in the MOX fuel, resulting in the need to process more spent PWR assemblies for each MOX assembly. Second, the presence of Np-237 provides a path to creation of Pu-238 through neutron capture, and as was shown in the MOX cases, the decay heat from increasing Pu-238 content partially offsets the benefit from transmuting the Pu-241. The extent to which neptunium recycle affects the overall benefit would need to be studied in a manner similar to this work.

Table 3. Comparison of the charge and discharge plutonium masses in each assembly type.

	UO2 MTIHM	MOX-1 MTIHM	CORAIL- PNA-1 MTIHM	IMF-1 Equivalent
Pu-238 charge	0.0	4072.0	232.5	2166
Pu-238 discharge	287.2	9000.0	759.9	4842
Pu-239 charge	0.0	83440	5991	44370
Pu-239 discharge	6007.0	57090	7714	2647
Pu-240 charge	0.0	39470	2808	20991
Pu-240 discharge	2653.0	37010	4243	10278
Pu-241 charge	0.0	16870	1166	8973
Pu-241 discharge	1952.0	17380	2597	6023
Pu-242 charge	0.0	11570	756.1	6154
Pu-242 discharge	887.7	12430	1704	9022

As would be expected from the previous work, it is the ability of each recycling strategy to control the relevant isotopes contributing to the decay heat and causing the temperature limit in the repository to be reached that defines the repository benefit for any strategy. This is accomplished by separating plutonium, americium, neptunium, cesium, and strontium from spent fuel, and sending the process waste to the repository. The plutonium, americium, and neptunium are recycled for further irradiation to transmute or consume these elements, and the largest repository benefits are achieved when only the process losses of these elements are ever sent to the repository. As shown by the results, the main load on the repository is from the direct disposal of the last assemblies in the recycling scheme, i.e., the burden on the repository in the MOX-1 case is almost entirely from the irradiated MOX-1 assemblies, not from process wastes.

VIII. Additional Comments

As additional comments, it is useful to note that processing and recycling spent PWR fuel could also be used to lower temperatures in a geologic repository, maintaining the reference drift

loading as currently planned for a repository at Yucca Mountain. This would allow more stringent temperature limits to be met, if needed, and could be used to provide greater assurance about the performance of the repository. Also, the reduction in the radionuclide inventory and the resulting alteration in estimates of the peak dose rate associated with releases from the repository have not been specifically addressed in this study, since this is not an issue for a repository at Yucca Mountain at this time as dose rate from releases does not constrain repository loading. However, some of the elements that are responsible for producing the decay heat in a repository (Pu, Am, Cs, and Sr), plus Np, are the main contributors to the dose rate. In a related ongoing study, quantifying the effects of the processing and recycling strategies using detailed repository performance assessment is showing that substantial dose rate reductions may be possible.

Systematic PWR fuel processing and recycling as evaluated in the current study would appear to offer other potential benefits for nuclear power, such as being able to densely dispose of process waste in a repository while greatly reducing the number of irradiated fuel assemblies that would need to be stored at reactor sites. Such possibilities emphasize the need to consider all relevant advantages and disadvantages when evaluating the usefulness of recycling in thermal reactors.

IX. Conclusions

The results described in this report for MOX, CORAIL-PNA, and IMF recycling strategies demonstrate that all of the thermal recycling strategies provide a benefit of increasing the maximum linear loading of repository drifts, with MOX providing the smallest improvement, followed by the CORAIL-PNA concept, and then the IMF approach. A summary of the benefits for each of the recycling strategies is shown on Figure 29 for processing 5 year old fuel, and in Figure 30 for processing 20 year old fuel (Figure 30 represents results from a separate AFCI work package that are included in for completeness of the repository benefits evaluations.). However, the largest benefit achieved is about a factor of 2.1, even with multiple recycling, which is far less than the factor of 40 or greater that might be expected, given the results presented in section I.A.

Results of this study have quantified the benefit to a repository as measured by increased drift loading of the repository; about a factor of 2 for the limited number of recycles considered. The results have also emphasized that approaches using a limited number of recycles will provide only a small fraction of the potential benefit (in excess of a factor of 40) that could be achieved with continuous recycling where the plutonium, americium, and neptunium remain in the fuel cycle, except for process losses. As discussed above, the general reasons for this have been determined as follows:

- In each case, the process waste is capable of being very densely loaded in the repository drift while still satisfying all thermal limits, about a factor of 40-50 greater than for spent PWR fuel. This is consistent with the previous results obtained for continuous recycling, since only process waste is sent to the repository in that case as well. (This can be increased to about a factor of 100 if curium were also separated from the process waste.)
- Direct disposal of the last assemblies in each recycling strategy requires most of the repository space, with allowable drift loading densities less than 5-10% that for spent PWR fuel. This is not unexpected, since all of the higher actinide elements from many assemblies have been concentrated in the assemblies containing recycled material.

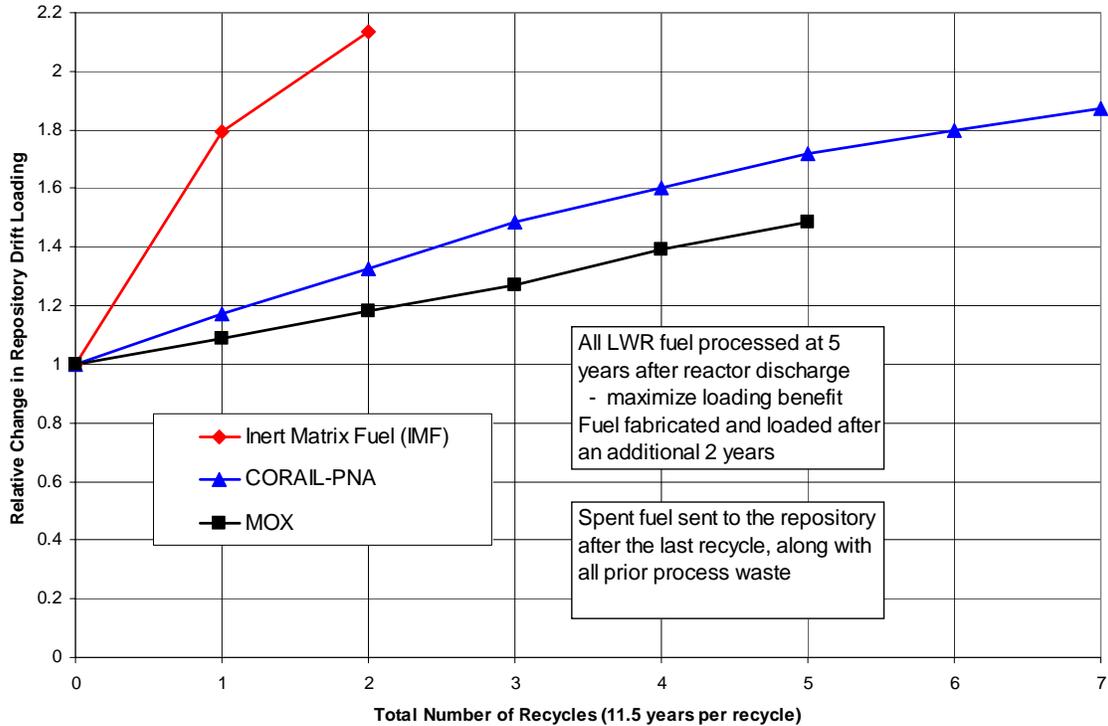


Figure 29. Summary of potential increase in drift loading with each of the three recycling approaches using fuel processed 5 years after discharge, normalized to an equivalent integrated energy basis

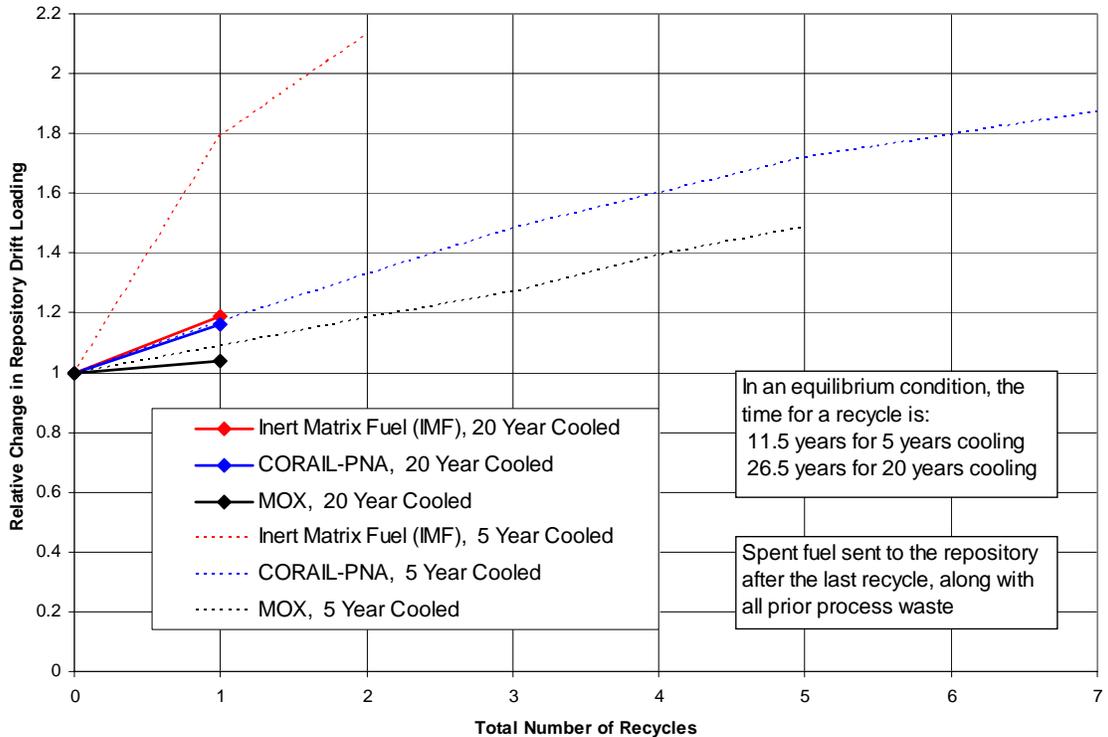


Figure 30. Summary of potential increase in drift loading with each of the three recycling approaches using fuel processed 20 years after discharge, normalized to an equivalent integrated energy basis

- It is the combination of the increased drift loading density for the process waste and the reduced drift loading density of the last assemblies in each case that limits the overall benefit to the loading of the repository to a factor of about 2.
- Continuous recycling is essential for obtaining large increases in drift loading in the repository, and may be possible in a thermal spectrum using the CORAIL concept, or advanced MOX, IMF, or target strategies. The key is to prevent a large fraction (>99%) of the heat-producing transuranic inventory from ever being placed in the repository.

Given these conclusions, it would be useful to further quantify the impact of recycling scenarios using thermal neutron systems, where production of the higher actinide isotopes is enhanced, as compared with alternate scenarios involving irradiation in a fast neutron spectrum, where production of higher isotopes is limited.

However, the results of this study should not be interpreted to mean that processing spent PWR fuel and recycling in LWRs is of little or no benefit:

- By processing the spent fuel and sending only process waste to the repository, substantial increases in loading (or corresponding decreases in repository size) are possible as soon as the activities are begun.
- As long as the recycled plutonium, americium, and neptunium are kept in the fuel cycle, the large allowable increase in drift loading, ~ 40 or more, is realized and maintained. The need to keep the plutonium, americium, and neptunium in the fuel cycle should not necessarily be viewed as a disadvantage, since in any version of a uranium-fueled nuclear future where nuclear power generation is at least maintained, the eventual deployment of fast reactors appears to be inevitable and they provide the eventual destination for the remaining plutonium, americium, and neptunium. It should also be noted that in the absence of a nuclear future, processing of spent nuclear fuel from existing reactors is likely to be unnecessary, as sufficient repository capacity is likely to be available even within a repository at Yucca Mountain to store all of the spent fuel.
- The use of LWRs for recycling, where they are essentially plutonium burners, reduces the number of fast reactors needed to provide the continuing recycling of higher actinides for a given number of LWRs, as shown in previous work.[6]

In summary, it has been shown that processing spent PWR fuel and using limited recycling in LWRs is of modest benefit to a geologic repository where loading is determined by thermal constraints. In addition, consideration of older fuel virtually eliminates any advantage from novel fuel types such as inert matrix fuel, and that use of mixed-oxide fuel provides essentially the same benefit. However, it has also been shown that recycling in LWRs can be very beneficial if it is part of an overall strategy where nuclear power generation continues into the future.

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