General Background Material

Workshop

Technical Basis

for

EPA HLW Disposal Criteria

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An Inside Look at the 40 CFR 191 Containment Requirements

by

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INTRODUCTION

Development of a system for the disposal of high-level and transuranic radioactive waste has proven to be one of the more difficult undertakings that U. S. society has confronted on the political, administrative, and technological fronts. Everyone involved has been criticized. Some critiques have suggested that we abandon the presently established system, from the law to the appointment of its implementers, and start over. It is understandable that the regulatory scheme should come under its share of criticism, and, we would sometimes say, more than our share.

The EPA standard (40 CFR 191), and especially its containment requirements, have often been a focus of these critiques. The fear is that a site and system, although offering very good protection, may be unable to pass these requirements, or at least it may be difficult to prove that it can meet the requirements. The identified sites potentially subject to this standard are still being characterized and no definitive answer as to their ability to meet the standard is possible. From time to time, situations are found, or analyses are conducted, that raise the question of whether a particular site will be able to meet the requirements. This creates great concern for those who see a successful waste disposal system as the last hurdle to the appropriate peaceful use of the atom.

We are equally interested in the early development of a successful waste disposal system, as long as it adequately protects the public health and the environment. We realize that it is not in the long range interest of those goals to have this material in temporary storage facilities for extended periods. This same philosophy has been expressed in other Agency programs by setting time limits for which chemically hazardous waste is allowed to be kept in storage. While such a prescriptive requirement is not appropriate for EPA's regulatory authority on high-level radioactive waste, it is no less a valid goal. It is also an issue that is directly addressed in the Nuclear Waste Policy Act.

We believe many of the criticisms of the EPA containment requirements have been made without an understanding of their purpose or the rationale behind them. This may be due to a lack of clarity in our explanation of the rule. In our redrafting of the rule, we have tried to improve both the description and the discussion of the underlying rationale. We hope this presentation will also help clarify some of the considerations that went into the development of the containment requirements. This should provide a better basis for future dialogue.

This presentation is not the place to recap the history of the EPA rule. It is necessary to point out, however, that while the standard was promulgated in 1985, Subpart B of the standard, which pertains to disposal, was remanded by a Federal Court in 1987. It is also germane that the containment requirements of Subpart B were not a part of the reason that the court ordered the remand. Because the court returned the whole of Subpart B, however, these provisions must be reproposed and justified anew.

THE CONTAINMENT REQUIREMENTS AND THEIR USE

The 40 CFR 191 containment requirements are not the easiest to explain or to understand. They differ from some of the radiation protection formulations of the past. Yet, when examined,

they do implement traditional radiation protection concepts that have long been fostered by both national and international bodies. Their difference stems from one predominant source: geological repositories that are expected to provide protection for thousands of years are different. These facilities are not appropriate for the day-to-day controls and monitoring that have characterized radiation protection in the past. They are not reactors with expected lifetimes of 40 to 60 years, and therefore they require a new form of standards to assure protection of the public and the environment.

The containment requirements in 40 CFR 191 are expressed as follows:

191.12 Containment Requirements

(a) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:

(1) have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix B); and

(2) have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix B)

(b) Performance assessments need not provide complete assurance that the requirements of 191.12(a) or (b) will be met. Because of the long time period involved and the nature of the events and processes of interest, there will inevitably be substantial uncertainties in projecting disposal system performance. Proof of the future performance of a disposal system is not to be had in the ordinary sense of the word in situations that deal with much shorter time frames. Instead, what is required is a reasonable expectation by the implementing agency, on the basis of the record before it, that compliance with 191.12(a) will be achieved.

In the Appendix C guidance section of 40 CFR 191, there is a statement of the Agency's intended procedure for demonstrating compliance with these containment requirements. This guidance, which is quoted in the box below, refers to the preparation of a "complementary cumulative distribution function". This is not a methodology that has been typically used in radiation protection. Therefore, it is appropriate to explain its usage further.

> Compliance with Section 191.12.
> The Agency assumes that, whenever practicable, the implementing agency will assemble all of the results of the performance assessments to determine compliance with 191.12 into a "complementary cumulative distribution function" that indicates the probability of exceeding various levels of cumulative release. When the uncertainties in parameters are considered in a performance assessment, the effects of the uncertainties considered can be incorporated into a single such distribution function for each disposal system considered. The Agency assumes that a disposal system can be considered to be in compliance with 191.12 if this single distribution function meets the requirements of 191.12(a).

The use of this technique can best be explained by referencing Figure 1 which is a graphical presentation of the format for showing compliance with the containment requirements.

Figure 1 displays a format where the probability of releases being exceeded defines the Y axis location while the amount of the release determines the appropriate plotting point on the X axis. The X axis, in this example case, is represented as multiples of the standard's Table 1 values. The curve formed by such a plot is known as a complementary cumulative distribution function (CCDF). The boundary criteria for compliance with the EPA containment

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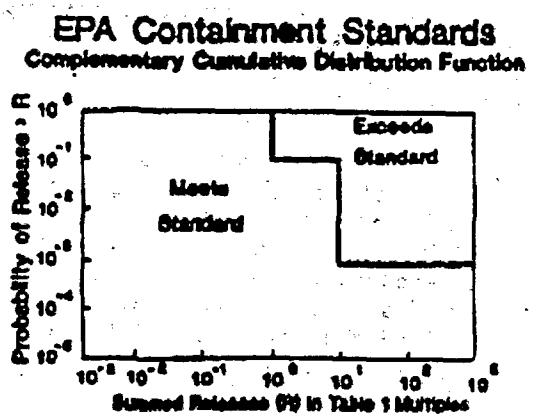

Figure 1 Graphical Representation of Containment Requirements

requirements are shown by the step function on the graph. If part of the CCDF representing reasonable expectaion falls to the upper right of this step function, the requirement is exceeded. If all of the CCDF falls to the left or below this step function, the requirement is met.

The step function is a graphical representation of the containment requirements of the standard. The Figure 1 step function also illustrates the EPA implementation guidance that performance assessments need not consider events or processes with less than one chance in 10,000 of occurring over 10,000 years. If, at a probability of 0.1 or greater the cumulative release of the events or processes exceeds the Table 1 values, the CCDF fails the test. If at a probability between 0.001 and 0.1 the cumulative release exceeds ten times the Table 1 values, the CCDF fails the test.

To create a CCDF for a disposal system, first identify the mutually exclusive events and processes that could lead to a release, define the probability of each, and estimate its consequences in terms of the curies released of various radionuclides. This identifies a set of scenarios and probabilities to use in plotting the CCDF. Using this base set of scenarios, with their probabilities and release consequences, you would start the CCDF plot with the highest consequence scenario with a probability greater than 10^{-4} and plot a point at its probability and release coordinate. Then moving to the next lower consequence scenario you would plot it at a coordinate representing its consequence and a probability point equal to its probability plus the probability of previously plotted events. The curve generated by this series of points represents the base CCDF for the disposal system.

Admittedly this is a very simplified discussion of the way compliance with the 40 CFR 191 containment requirements is expected to be determined. There are many qualifiers to be considered. It is not the sort of thing you generate in a short period. It is expected to be developed with sophisticated computer analysis systems. Nevertheless, this description should allow the uninitiated to understand the subsequent discussions of purpose, development, and implemeatation.

THE PURPOSE OF CONTAINMENT REQUIREMENTS

The containment requirements serve several purposes. The first of these is as a measure of the integrity of the repository. This was a prime consideration in their development, and still remains a major reason for retaining this form. At the time of their initial development we were in the mode of comparing several potential repositories as to how well they could retain the waste. The most obvious approach was to compare the total releases over some period of time.

This comparison of releases was intimately linked with the time frame that was chosen for the containment requirements. We wanted a time long enough to see differences among different geologic media. Over periods of hundreds, or even a few thousand years, all geological repositories look pretty much alike as to their total releases. Ten thousand years of modeling generally allows enough time for differences to begin to show up. It is also an adequate period to discern a very good estimate of the integrity of a single repository site, as in the present U. S. situation. Ten thousand years is also short enough to avoid getting into endless debate over things such as when the next glaciation will occur.

A second, and no less important, purpose of the containment requirements is they provide a direct correlation to the number of premature health effects that might be expected, a term referred to as the detriment. For standard setting purposes, we assume that all radiation exposure contributes some proportional amount of risk, and that the health effects are proportional to the product of the number of people exposed and the amount of dose they each receive. This is a perspective that is not available when we only look at the exposure of individuals. If only individual exposure were to be limited, we might be led to seek dilution as a way to avoid exceeding a standard. The use of total releases in the form of the containment requirement avoids this pitfall.

The last containment requirement purpose we will discuss is the advantages of the assessment and analyses that it forces one to conduct. Because of its probabilistic nature, it forces the assessment of both natural and man-made *disruptive* events. These disruptive events can be the most significant of the possible release mechanisms. On the other hand, their inclusion receives a good bit of criticism because they are the most difficult to analyze. We acknowledge this difficulty, and have made provisions for handling uncertainty and using expert opinion to develop estimates in areas where a historic base is inadequate. We believe it would be negligent not to include these types of release mechanisms in the assessment. From the work we have seen on actual initial assessments, we are further persuaded that some of the potential problems uncovered, and now receiving intensive investigation, might never have been addressed if it hadn't been for the form of these containment requirements and their probabilistic features.

THE DEVELOPMENT OF THE CONTAINMENT REQUIREMENTS

The development of the containment requirements was based on a generic assessment of the protection provided by well sited, well designed geologic repositories. This assessment was based on a repository that would contain 100,000 metric tons of heavy metal (MTHM) or its equivalent. This is about the amount of spent fuel that would be expected to be produced by 100 nuclear power reactors of current design over their operating life.

In the generic analysis we evaluated several geological media including salt, basalt, and tuff. The environmental pathway modeling was very simplistic as befits a generic assessment. Under the Atomic Energy Act Authority the Agency may only develop its standards as "Generally Applicable Environmental Standards." This precludes making them either site specific or specific as to their method of achievement.

The pathway model used for undisturbed releases applied classical transport mechanisms, including appropriate media retardation, to move any material that escaped from the confines of the repository, through the ground water, and into surface waters of the general environment. Radionuclides were considered available for interaction with man through consumption of water and irrigated food. Other peripheral pathways were also evaluated. These included fish consumption, and the consumption of animal products which had fed on contaminated pasture and irrigated crops. Actions that would disrupt the repository, whether of a human or natural origin, were considered as initiating events that could cause a release from the repository into the pathway just described.

The one unique aspect of the modeling was avoiding detailed assumptions about future human population habits and distribution relative to the surface waters of concern. It has been found that the ratio of population to river flow tends to a constant value world wide. The assumed population for the futuristic world within the period of analysis (10,000 years) was assumed to be roughly double that of today or ten billion people. The world annual river flow is about 3×10^{16} liters per year. The ratio of population to river flow that was used is then 3.3×10^{-7} person-years per liter. This value was then used for all pathways where the radionuclides in surface water were the source of human exposure. The world average value used compares to a U.S. average of 1.3×10^{-7} and individual river range of 2.4×10^{-6} to 5.7×10^{-7}. It was also assumed that the basic habits and consumption rates were the same as today's

population.

Based on these generic intakes and dosimetry health effects estimates were made. A linear, non-threshold relationship between radiation exposure and premature fatal cancers was assumed. This means we assumed that any additional exposure added some increment of cancer risk and that this risk was proportional to the dose received. This has long been the generally accepted prudent approach for risk assessment when used as a basis for setting radiation protection standards. The conversion factor from dose to health effects that is currently being used is an annual risk of about 4×10^{-7} fatal cancers for each person-millirem. A lifetime exposure duration of approximately 71 years is used which results in an individual lifetime risk of about 2.8×10^{-5} fatal cancers for each millirem of annual exposure.

The one exception to this general approach of exposure pathway analysis was for the radionuclide carbon-14. After consultation with a special radiation subcommittee of the Agency's Science Advisory Board, it was determined to be more appropriate to use the specific activity approach as developed by G. G. Killough of the Oak Ridge National Laboratory. This methodology evaluates the exposure from carbon-14 based on the incremental increase that the release will cause in the world's ratio of carbon-14 to stable carbon. Since any carbon released from a repository can be assumed to generally come to equilibrium with the world's carbon cycle during the 10,000-year period of analysis, it is possible to determine the population exposure using a single ratio. For a world population of ten billion we arrived at a ratio of 399 person-rem per curie of carbon-14 released.

There are certain limitations to the EPA modeling methodology. As previously mentioned, the models were generic and would not be suitable for a site specific evaluation. This is appropriate to the EPA authority. Site specific evaluations would be expected to utilize site specific data based on the site characterization.

Another limitation is the EPA assessment did not consider the possibility of undisturbed repository gaseous releases. At the time the initial analysis was done, gaseous releases were not considered credible by the Nuclear Regulatory Commission nor the Department of Energy. This has been a matter of recent attention, since it is now perceived that a very porous unsaturated site may allow gaseous releases. However, the modeling of releases from an unsaturated site is a highly uncertain issue and it is difficult to determine all the involved parameters and their effect. In our view, this is an area still requiring much study and evaluation by all the involved agencies, both as to the possible extent of such releases and their potential mitigation.

From these generic analyses the Agency found that deep geologic disposal could provide extremely good protection from releases of radioactive material to the accessible environment. In comparison with other impacts on man from both natural events and man's activities, releases from high-level and transuranic radioactive waste repositories only resulted in few health effects (tens to hundreds of premature fatal cancers over 10,000 years). However, it was also realized that these generic assessments did not consider the site specific factors, which contribute to uncertainty. Based on the analysis of a repository's capability, and considering the uncertainty factors, the Agency determined that a very limited number of health effects should be allowed for repository releases and set that value at 1,000 premature cancer deaths over the 10,000-year period of analysis (an average of less than one premature cancer per year).

Several other considerations assured the Agency that this was an appropriate level of protection. One of the basic philosophies established by advisory bodies, such as the International Atomic Energy Agency, is that waste disposal activities should not impose any greater impact or burden on future generations than was acceptable to our own generation. The limitation of 1,000 premature fatal cancers in 10,000 years appears to meet this criterion since conservative estimates of the impact from the nuclear fuel cycle have predicted an average of around 100 premature fatal cancer deaths in only the next 100 years. This estimate of the fuel cycle risk comes from the paper by W. H. M. Ellett and A. C. B. Richardson titled Estimates of the Cancer Risk Due to Nuclear Electric Power Generation and

Published in *Origins of Human Cancer* © 1977 Cold Spring Harbor Laboratory.

For perspective, the Agency also made an assessment of the potential impact that would have resulted from an equivalent amount of unmined uranium ore if it had never been removed to produce nuclear fuel. Using the same basic model used to assess repository performance, we found that population risks from this uranium could be between 10 and 100,000 premature cancer deaths over a 10,000-year period depending on the natural variability of the formations and the analytical assumptions made. The selected limit for a repository falls well within this range further assuring that this is an acceptable level of protection.

The analysis was also reassuring from the standpoint of the projections over time. Although the base analysis was limited to 10,000 years, we examined, in a more qualitative way, what might happen for longer time periods We found if the goal of the standard was met for 10,000 years, there was no basis to speculate that significantly different releases would occur immediately beyond that time. We believe this should be confirmed on a site-specific basis. Although the increased uncertainty of such longer-term extrapolations make them unsuitable for quantitative analyses in a regulatory process, this look over the longer time horizon is helpful in providing perspective.

We then used this level of protection (1,000 premature fatal cancers in 10,000 years) as the basis for calculating the release limits specified in Table 1 of the standard.

To select the specific release limits for the various radionuclides we used the same models to estimate the health effects that might be caused by their individual release. The release limits in Table 1 were then calculated as to how many curies of each radionuclide would cause 1,000 premature deaths over 10,000 years if released to the environment. The limits were then stated in terms of the allowable release from 1,000 metric tons of reactor fuel (therefore the actual curie values in Table 1 correspond to a risk level of 10 premature deaths over 10,000 years). Because of the approximate nature of these calculations these limits have generally been rounded to the nearest order of magnitude based on the logarithmic midpoint of the range. Release limits for disposal systems will be based upon the amount of waste in the system. If, for instance, a disposal system is ultimately used to dispose of 70,000 MTHM, the release limits for the facility would be the limits of Table 1 times seventy (70,000 MTHM divided by 1,000 MTHM).

For some of the wastes covered by this rule, 1,000 metric tons of reactor fuel is not an appropriate unit of waste. In these situations, the Notes to Table 1 provide instructions on how to calculate the release limits. For example, this is the case for high-level wastes from national defense applications which contain much different amounts of radioactivity.

IMPLEMENTATION

Mixtures of Radionuclides

In the performance assessment required by the containment requirements of 40 CFR 191, it is not expected that there will be only one radionuclide released. Rather, the analysis will likely show that a number of radionuclides will be released in various amounts. Since each of the values in Table 1 constitute a release resulting in approximately 1,000 premature fatal cancers, it is inappropriate to measure each against the full table value. The appropriate technique to use is the sum of the fractions as described below:

For each radionuclide in the release mixture, determine the ratio between the cumulative release quantity projected over the 10,000 years and the limit for that radionuclide as determined from Table 1 and its associated notes. The sum of such ratios for all the radionuclides in the release mixture may not exceed one with regard to 191.12(a)(1) and may not exceed 10 with regard to 191.12(a)(2).

These requirements are illustrated in the following equations: If radionuclides A, B, and C are projected to be released in amounts Q_A, Q_B, and Q_C and if the applicable release limits are RL_A, RL_B, and RL_C then the cumulative releases over 10,000 years shall be limited so that for 191.12(a)(1) the relationship of Equation 1 exists:

$$\frac{Q_a}{RL_a} + \frac{Q_b}{RL_b} + \frac{Q_c}{RL_c} \leq 1 \qquad \text{Equation 1}$$

Similarly for 191.12(a)(2) the relationship of Equation 2 exists:

$$\frac{Q_a}{RL_a} + \frac{Q_b}{RL_b} + \frac{Q_c}{RL_c} \leq 10 \qquad \text{Equation 2}$$

Uncertainty and Quantile Considerations

In preparing a CCDF, there will be uncertainty in the parameter values, events, and processes that make up the scenarios that are included. Although the EPA containment requirements, and their associated guidance in Appendix C, clearly indicate that it is sufficient to resolve the matter into a single CCDF that represents the implementing agency's evaluation of "reasonable expectation", it is often useful to represent the analysis as a family of CCDFs as is shown in Figure 2.

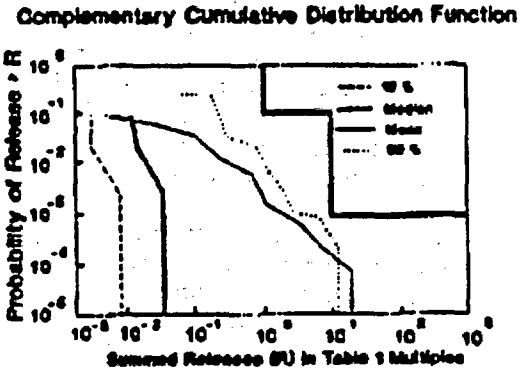

Figure 2: Family of CCDFs at Different Uncertainties

Figure 2 depicts CCDFs representing the 10% and 90% quantiles, and the mean and the median. It is interesting to note that in the case shown in the figure the mean exceeds the 90% quantile at the lower probabilities. This is not as unusual as one might at first think, and results from one or a few release scenarios that have far greater consequences than the others. This illustrates that it may not be appropriate to use any one specific CCDF member of the curve family as a universal criteria of reasonable expectation. Certainly we would anticipate that something greater than the median would be considered for this criteria.

Human Intrusion

One of the aspects of the containment requirements that has received much discussion is their inclusion of human intrusion as one of the potential disturbing events to be considered. The major difficulty with human intrusion is in predicting appropriate probabilities for the events. Because of this, some have suggested separating out the human intrusion analysis in some way. We have examined this possibility and have been unable to find a way to do this that would, in the end, allow us to represent the total potential risk from the repository. We also looked at the possibility of using a deterministic approach to the human intrusion analysis, but this only gives the analyst two choices for a release scenario, either it happens or it does not. Since it is very difficult to prove that something will not happen, and this seemed to require an even more absolute probability determination than the containment requirements, we have not followed this approach.

Because we realize the difficulties with human intrusion analysis, we have defined some limiting boundaries on what the analysis must include. First, it must only consider *inadvertent intrusion*; purposeful intrusion, whether it be for recovery of the materials in the repository, or for some act of sabotage, are not to be considered. If a future generation decides to go into the repository for some material that we discarded there, we believe that it is their responsibility to evaluate the risks to themselves and future generations. Sabotage is no more predictable than speculating on whether some future generation, in the heat of war, will detonate a nuclear device on the repository. We believe such an event is not a part of the analysis that is appropriate here. We believe that the attention of the analyst should be directed to those types of intrusion that have some possibility of mitigation through site selection, engineering design, and appropriate institutional controls. One of the most evident measures, which is also in the assurance requirements of 40 CFR 191, is to locate the repository away from underground

resources that man is likely to want to recover. Other measures include engineered barriers, waste form, and waste spacing.

Institutional controls that may be considered are of two types, active and passive. Active institutional controls require man to maintain them or to take some other type of action. This would include such measures as guards, fences, and preventive maintenance. Passive institutional controls are provisions like government ownership, monuments, records, and other devices intended to warn future generations to avoid entry into the repository.

The EPA standard allows active institutional controls to be considered as mitigating the probability of intrusion for the first 100 years after repository closure. The 100 years was the result of extensive consultation with scientific bodies, including the Agency's Science Advisory Board, and discussions at public forums. Longer times were considered, but instances like Love Canal were too prevalent to justify them.

Passive institutional controls can be considered as mitigating the probability of inadvertent human intrusion as long as it is not given credit for 100% assurance. Several other aspects have been included in the implementation guidance. The analyst can assume a state of intelligence in future generations that is at least equal to that of today. In this context he can assume that if future man does intrude he has the capability to realize what he has done and to take appropriate action. The specific credit given to various passive institutional controls is left to the repository developer and the implementing agency for compliance. The EPA standard includes in its Guidance for Implementation section values of drill hole density and hole sealing integrity that we would consider the worst that need be considered. It is anticipated that site specific evaluations will determine and support more appropriate values to use in the actual analysis.

Uncertainties

The handling of uncertainties in preparing a CCDF has been a matter of much discussion. With the realization that this required a number of considerations unique to this form of a standard, EPA included a number of guidance statements to clarify our intention. For instance, in discussing how the implementing agencies might assure compliance where predictions of performance are made, we said:

> Substantial uncertainties are likely to be encountered in making these predictions. In fact, sole reliance on these numerical predictions to determine compliance may not be appropriate; the implementing agencies may choose to supplement such predictions with qualitative judgments as well.

Also, to assure that there is appropriate truncation to the probabilistic analysis, we included the following guidance:

> The Agency assumes that such performance assessments need not consider categories of events or processes that are estimated to have less than one chance in 10,000 of occurring over 10,000 years. Furthermore, the performance assessments need not evaluate in detail the releases from all events and processes estimated to have a greater likelihood of occurrence. Some of these events and processes may be omitted from the performance assessments if there is a reasonable expectation that the remaining probability distribution of cumulative releases would not be significantly changed by such omissions.

Because it was clear that parameter uncertainties were very much a part of the analysis, we included some thoughts on how they might be handled with guidance that:

> When the uncertainties in parameters are considered in a performance assessment, the effects of the uncertainties considered can be incorporated into a single such distribution function for each disposal system considered. The Agency assumes that a disposal system can be considered to be in compliance with 191.12 if this single distribution function meets the requirements of 191.12(a).

Individual Annual Dose versus Total Release

Probably the area in which we have had the most consistent difference with the various advisories is that of probabilistic-related total release assessment. Although we have set individual annual exposure levels for the undisturbed performance, we have taken the approach of setting limits on total releases over 10,000-years for the probabilistic-related standards. When we started this standard setting effort, it was our inclination to use individual dose, since that was how radiation standards had always been set. It was only after we examined what it would mean to have to comply with such a provision that we switched to our present approach. We believe that approach is much more appropriate in view of the long time periods and uncertainties involved. The easiest way to show why we came to this decision is through reference to Figure 3.

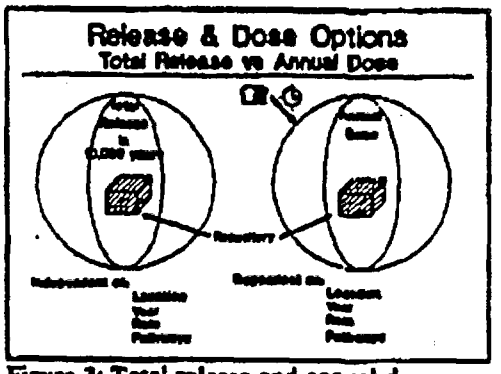

Figure 3: Total release and annual dose alternatives

Figure 3 shows two symbolic spheres representing the boundary line around two repositories and the defined "accessible environment". The sphere on the left represents the compliance case if you have the current structure of the EPA standards for probability related releases. Under these circumstances, one only needs to estimate the probability and quantities of releases of radioactive material across this boundary at any time during the 10,000-year period. To demonstrate compliance, it is not necessary to identify where on that boundary sphere the release occurs (Location), when the release starts or stops during that 10,000-year period (Year), the time-related frequency of the amounts released (Rate), or how this material might interact with people and how they might be exposed (Pathways).

If we were to pursue the second alternative, annual individual dose, as depicted by the sphere on the right of Figure 3, we have a much more difficult analytical task. It is no longer sufficient to just estimate how much radioactivity is likely to cross the boundary. We now must also estimate where (arrow) on that boundary it will occur and whether it is close to people. That would be only the beginning of the analytical chore, however. Since we must calculate an annual dose, we must know when the release starts and ends (clock) and how much will be released on an annual basis (time-related factors on an annual basis). While we think all this is more than should be asked of a probabilistic analysis, it would not be sufficient. We would still be required to go on from that point and speculate on how this material might interact with people (pathway-bread) at this specific location and what the subsequent annual dose might be. These are speculations we believe are feasible for the analysis of the undisturbed repository but that are beyond what should be considered reasonable for the probabilistic analysis.

CONCLUSION

In order to promote meaningful discussion of the Containment Requirements in EPA's standards for high-level and transuranic wastes, we have presented an in-depth discussion of their many aspects. It is our belief that these requirements fulfill purposes which are uniquely appropriate for these waste materials. Their development and implementation considerations are multifaceted and intertwined. If changes are to be considered to their form, or content, the possible effects of all these aspects need to be evaluated. ■

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AT THE

SYMPOSIUM ON RADIOACTIVE WASTE REPOSITORY LICENSING
NATIONAL ACADEMY OF SCIENCES
NATIONAL RESEARCH COUNCIL

WASHINGTON, D.C.
SEPTEMBER 17, 1990

"REPOSITORY PERFORMANCE -- THE REGULATORY CHALLENGE"

INTRODUCTION

Good morning, ladies and gentlemen. It is a pleasure and indeed a distinct honor for me to have this opportunity to deliver the keynote address this morning, opening what I consider to be a most timely and important symposium on a subject of great interest to all of us, the licensing framework for radioactive waste repositories.

This Symposium comes at an important time in the waste disposal program, with the regulatory agencies -- the Environmental Protection Agency (EPA) and we at the Nuclear Regulatory Commission (NRC) -- currently focusing our attention, respectively, on EPA's ongoing efforts to establish the basic health and safety framework that will govern the licensing of a geologic repository and, following that, on NRC's efforts to conform our more detailed licensing requirements to those EPA standards.

In July of this year, the Board on Radioactive Waste Management of the National Research Council released a report entitled "Rethinking High-Level Radioactive Waste Disposal", which examines a wide range of issues concerning the licensing process for geologic repositories. This report, together with a number of other recent analyses and commentary, have served to focus increased attention -- and, in some quarters, concern -- on the regulatory standards and process for the licensing of a geologic repository. It is this subject that I would like to focus on in my remarks here this morning.

In so doing, however, -- and particularly in view of my training as a lawyer and not a scientist -- what I propose to do is to step back from many of the detailed technical issues that occupy those who work in this arena on a day-to-day basis -- issues such as APEs and UPEs, CCDFs, and the like --and instead focus on what I think are the broader policy questions that have arisen with regard to the regulatory framework, beginning first with an examination of EPA's approach to establishing a generally applicable environmental standard; then focusing on the relationship between EPA's standard and NRC's implementing regulations -- what I will refer to as the technical nexus; and then, finally, turning my attention to the approach taken by the NRC in its implementing regulations and, specifically, the approach that we have taken to subsystem performance criteria.

Perhaps before I begin -- and again, reflecting my own personal background and biases -- let me turn first to the Nuclear Waste Policy Act for a brief but important reminder of what the statute itself says about the regulatory framework for geologic repositories: Section 121 of the Act directs EPA to establish generally applicable standards for protection of the general environment from offsite releases from radioactive material in repositories; that same section goes on to direct the NRC to establish the detailed technical requirements, including a system of multiple barriers, for the licensing of such repositories.

Importantly, the approach used here was one that Congress had used elsewhere: EPA was to set the generally applicable environmental standards, reflecting their conclusion as to what would be required to protect the public health and safety; and NRC, in turn, was to establish the detailed technical requirements, defining how, specifically, one would go about meeting EPA's general standards.

I emphasize this distinction not simply because it exists, but because it is an important one -- and one that I will return to when I discuss the relationship between EPA's standards and NRC's implementing regulations.

EPA STANDARDS

With that by way of general background, let me now turn to the first of the three issues that I would like to address this morning, the subject of EPA's generally applicable environmental standard. As I indicated at the outset of my remarks, EPA is currently in the midst of repromulgating its general standard, responding to a court remand of July of 1987.

It is in this context that the opportunity has arisen, once again, for those within EPA who are responsible for the development of this standard, as well as those of us outside the agency who have an interest in this matter, to reexamine not just that part of the standard that was the subject of the court remand, but to consider as well the entire range of issues, including the containment requirements, that have been so controversial over the years.

Two issues, in particular, have arisen as a result of the pending review: First, the overall stringency of the EPA standard; and second, the feasibility of implementing the probabilistic portion of the standard. Neither of these two issues, of course, is new and, for that reason, I don't propose to review the lengthy history associated with either one here this morning.

But what I do think is important here and what ought to be emphasized is the continuing and unresolved debate over both of these issues. Indeed, it was this very point that the NRC made in our letter of August 27, 1990, commenting on EPA's Working Draft Number 2. With regard to the stringency of the proposed standard, the letter observed, and I quote:

> "There continues to be considerable controversy regarding the stringency of [EPA's standards]. This controversy results, at least in part, from concerns over the very low levels of risk which underlie release limits imposed by EPA's standards, particularly when compared to other federal health and safety standards. We are concerned that a clear understanding and acceptance of the standards will not be achieved until EPA has explicitly documented the acceptable risk level that underlies the release limits of the standards and the way in which the release limits were derived from that risk level."

With regard to the ability to implement the EPA standards, the NRC letter of comment indicated that:

> "[T]he Commission continues to be concerned about the workability of standards that require numerical probability estimates for very unlikely processes and events."

In my judgment, the debate over the stringency of the standards will continue, perhaps notwithstanding whatever explanation EPA might provide. Nevertheless, as the Commission emphasized in its comment letter, the basis for the proposed standards should be thoroughly documented and, more importantly, explained in the context of regulatory standards that have been imposed for other risks experienced by society, so that those who might wish to evaluate the stringency of the EPA standard will have a framework within which to do so.

Of greater concern, at least in my personal judgment, is the continuing controversy over the ability to implement and apply probabilistic standards as a basis for licensing a geologic repository. This issue, as I indicated, is not new either. Indeed, it was first raised by the Commission in 1978, when the Commission, in commenting on this matter to EPA, said, and I quote:

> "We feel strongly that a deterministic method should be used to regulate nuclear facilities. We are aware that you are considering a substantially different type, a probabilistic standard which requires quantitative risk assessment. Based on our understanding of the virtues and the weaknesses of quantitative risk assessment, we are convinced that it can and should be used to provide insight on the quality and effectiveness of HLW disposal regulation, but it cannot be the explicit basis of the regulation . . ."

The discussion of this most important issue continued over the next several years, culminating in EPA's promulgation of its standard -- with its basic probabilistic framework intact -- in September of 1985. Importantly, the standard included the following proviso:

> "Performance assessments need not provide complete assurance that the requirements of [this standard] will be met. Because of the long time period involved and the nature of the events and processes of interest, there will inevitably be substantial uncertainties in projecting disposal system performance. Proof of the future performance of a disposal system is not to be had in the ordinary sense of the word in situations that deal with much shorter time frames. Instead, what is required is a reasonable expectation, on the basis of the record before the implementing agency that compliance with [the standard] will be achieved."

It was based upon the inclusion of this language in the final EPA standard, that the staff advised the Commission at the time that EPA's probabilistic standards can, indeed, be implemented in a licensing review.

Four years later, in October of last year, the staff came back to the Commission, once again raising the implementation issue -- this time in the context of EPA's now-ongoing response to the court's remand. Acknowledging that EPA had endeavored to address this issue in its 1985 standard in the fashion recommended at the time by the NRC, the staff noted this past October that:

> "[W]hile the language added by EPA to the rule and in the Supplementary Information [in 1985] tends to recognize qualitative considerations, an acceptable approach to implementation is still ambiguous and the governing standard is still the probabilistic numerical standard."

Thus, we find ourselves today -- nearly 12 years after the NRC first identified the difficulties associated with implementation of a probabilistic standard in a licensing context -- facing what I consider to be a most unfortunate situation: We have yet to identify a clear and unambiguous approach to implementing EPA's standard in our licensing review.

What we do about this situation, of course, is the challenging question -- and with your forbearance, this is a topic that I intend to address shortly, when I turn my attention to the NRC regulations and their relationship to the EPA standards. I would emphasize at this point, however, that if the EPA standards are to serve as an independent regulatory basis for licensing the repository, a clear and unambiguous resolution of this issue is essential and urgent.

RELATIONSHIP OF NRC REGULATIONS TO EPA STANDARDS

Lest it appear that the EPA standards are the source of my greatest concern, let me now turn my attention to the remaining two issues that I would like to discuss this morning -- both of which more directly involve our approach at the NRC to establishing detailed implementing regulations -- beginning first with the relationship between EPA's standards and NRC's regulations.

As I indicated at the outset, the Nuclear Waste Policy Act envisions that EPA would establish generally applicable environmental standards, setting forth the general offsite limits necessary to protect the public health and safety, with the NRC in turn establishing the detailed technical requirements necessary to achieve those standards.

This approach, as I indicated, is one that Congress had used in other contexts -- and, specifically, in the Uranium Mill Tailings Radiation Control Act of 1978. In addition, it draws directly upon the authority that was transferred to EPA, when that agency was

first established, to promulgate generally applicable environmental standards.

In view of the division of responsibility between the respective agencies -- EPA setting the overall standards, NRC establishing the implementing regulations to achieve those standards -- you can imagine that it came as quite a surprise to me when, in the context of a recent Commission meeting -- where the Commission and the staff were discussing the difficulty of demonstrating compliance with EPA's probabilistic standard in our licensing proceeding -- I learned for the first time that one did not necessarily comply with the EPA standards by demonstrating compliance with the NRC regulations. Indeed, as it turns out, the same result attaches to the converse: one does not necessarily comply with the NRC regulations by demonstrating compliance with the EPA standards.

While it is true that compliance with the NRC regulations makes it more likely that the applicant will satisfy the EPA standards, the technical nexus that I was searching for -- and that, in my judgment, the concept of a generally applicable environmental standard contemplates -- was not sufficiently well-established to permit one to reach the legal conclusion that compliance with the NRC regulations could be deemed to constitute compliance with the EPA standards.

The advantage of such a nexus is obvious: One need not litigate the underlying EPA standard, with all of the attendant difficulties associated with its probabilistic character, if the deterministic NRC regulations were based upon a clear technical nexus to the EPA standard. But unfortunately, that is not the situation that we have today.

One can quite readily understand, historically, why this technical nexus between EPA's standards and NRC's regulations does not exist -- recall that in the late '70s and the early '80s, NRC was endeavoring to convince EPA of the problems associated with a probabilistic standard and, indeed, had its own deterministic standard out on the streets, in final form, before the EPA standards were promulgated. Unable to convince EPA to abandon the probabilistic approach when it finalized its standards in 1985, we now have what amounts to two legally distinct licensing standards -- NRC's deterministic requirements in 10 CFR Part 60, and EPA's probabilistic standards in 40 CFR Part 191.

I recognize, as the staff concluded in NUREG-0804 and its subsequent appendices, that compliance with NRC's regulations does indeed make it <u>more likely</u> that EPA's standards will be met. However -- and perhaps here I view this issue more through the eyes of a lawyer reflecting upon the challenge that the applicant will face in a licensing proceeding -- absent a technical nexus that would permit one to conclude that compliance with NRC's regulation can be deemed to constitute compliance with EPA's standards, I fea

that the applicant will face the challenge of demonstrating that both the NRC regulations and the EPA standards have been satisfied.

In short, while we may have reduced the technical uncertainty with such an approach, we have increased the legal uncertainty, perhaps to the point where the various requirements that the applicant must demonstrate compliance with in a licensing context -- and particularly those that involve probabilistic elements of proof -- will overwhelm whatever technical advantage might otherwise exist in such an approach. Indeed, as Pogo once observed, we may be facing insurmountable opportunities.

What is the solution? One obvious answer would be for the NRC to reevaluate its subsystem performance criteria, following promulgation of the EPA standard, with an eye towards restructuring those criteria in a manner that would permit one to establish the technical nexus that I referred to earlier. We already know, for example, that one of the three subsystem performance criteria -- the 1,000 year groundwater travel time --is not only a poor surrogate for radionuclide transport, but moreover, does not appear to have a clear nexus to the EPA standard and, for this reason, may not be necessary in 10 CFR Part 60.

I suggest such an approach recognizing that the conclusion of an analysis such as this may well be that the subsystem performance criteria should be made more stringent -- but with the caveat that if this pushes these criteria beyond the bounds of what is technically achievable, then it seems to me that this indirectly raises a question about the stringency of the underlying EPA standard -- which, you recall, was based upon a view at the time as to what could be feasibly achieved. In the event that such a situation comes to pass, this, it seems to me, is a matter that we should then pursue directly with EPA.

The benefit of such an approach, I emphasize, is that the licensing framework for the geologic repository would be based upon deterministic considerations, rather than the very difficult and perhaps intractable considerations that we would face in applying -- and litigating -- a probabilistic licensing standard.

NRC SUBSYSTEM PERFORMANCE CRITERIA

With the time remaining, let me now turn to the third topic that I'd like to cover this morning -- the specific approach that we at the NRC have taken in establishing subsystem performance criteria for the repository. And here what I would like to focus on -- as distinct from the relationship of our regulations to EPA's standards -- is the somewhat narrower question of the approach that we have taken to establishing specific subsystem performance

criteria and the flexibility contained in the application and implementation of those criteria.

As I reviewed the report "Rethinking High-Level Radioactive Waste Disposal", one of the more insightful comments of that report, in my judgment, was its call for a certain amount of flexibility in the implementation of regulatory requirements. And indeed, it is in the context of the NRC's subsystem performance criteria that this recommendation, in my view, appears most appropriate.

As many of you know, we currently have three subsystem performance criteria, set forth in 10 CFR Part 60, each of which represents a minimum requirement and each of which must be satisfied as a prerequisite to issuance of a construction authorization: a release rate for any radionuclide from the engineered barrier system of one part in 100,000; a groundwater travel time criterion of 1,000 years; and a package lifetime criterion of 300 to 1,000 years.

I emphasize, each of these criteria constitutes a minimum requirement and each must be satisfied independent of the others.

That, of course, raises the question as to whether, for example, we would be satisfied with an application that relied to a greater degree on one of the criteria -- say, package lifetime, for instance -- and, based upon that enhanced performance, we would perhaps take that into account in evaluating compliance with any of the other criteria.

I emphasize here that the regulation specifically authorizes the Commission, on a case-by-case basis, to approve some other radionuclide release rate, designed containment period, or groundwater travel time, "provided that the overall system performance objective is satisfied."

But there are two aspects of this provision -- which requires an affirmative Commission decision to implement -- that I would like to emphasize here. First, as I indicated, this provision requires an affirmative decision by the Commission, in order to permit the applicant to depart from the specific subsystem performance criteria defined in the regulation; rather than allowing the applicant, on its own initiative, to determine how best to strike the balance between these three subsystem performance criteria and then to demonstrate that the overall level of protection sought by the Commission has been provided.

Purely as a practical matter, I think it highly unlikely that as we get closer to the licensing process, with the intense public scrutiny that will no doubt exist at that time, the Commission will be in a position to approve the kind of tradeoff that might make sense -- to take one example, greater reliance on the engineered waste package, with perhaps lesser reliance on groundwater travel

time -- if that results in the applicant being allowed to meet a less demanding showing on any of the criteria. This is particularly true, in my judgment, when one recalls that these individual subsystem performance criteria were based upon a conclusion as to what is technically feasible, not necessarily what is required to protect the public health and safety. Are we to allow DOE, the argument will go, to do less than what is feasible?

Beyond the practical problem, the second concern that I have with our ability to apply these criteria in the flexible manner that I think was originally intended is that these criteria were not based upon some well-defined overall health and safety goal -- established either in the body of NRC's regulations or, alternatively, in EPA's general standards. Indeed, as I just indicated, these criteria reflect an agency judgment as to what is feasible, not necessarily what is required to protect the public health and safety. Thus, whether DOE is allowed the flexibility to strike its own balance or NRC is required to take some affirmative action to approve an alternate approach, the problem that we have in either case is the same: How do we define the overall system performance objective -- the standard for evaluating any alternative approach that DOE might propose --when -- (i) an unambiguous technical nexus to the EPA standard is lacking; and (ii) the subsystem performance criteria were formulated to reflect that which is feasible, rather than with an eye toward an overall safety objective?

Here, it seems to me that the answer to this problem, again, lies in establishing a firm technical nexus between the NRC regulations and the EPA standard -- and then allowing the applicant, without the need for affirmative Commission action, to decide what emphasis to place on individual subsystem performance criteria, so long as the Commission determines that the overall performance objective -- ideally the EPA standard -- is satisfied. Indeed, I think you will find that this approach is very much akin to the philosophy reflected in the Commission's recently-approved implementation plan for applying the Safety Goal to commercial nuclear power plants, where less emphasis has been placed on specific subsystem criteria, so long as the overall safety objectives established in that policy are satisfied.

CONCLUSION

Let me conclude my remarks with the following observation: It is my hope that the remarks that I have delivered here this morning will serve to stimulate a healthy discussion over the next two days about how we might improve the regulatory process for licensing a repository in this country. I do not count myself among those who believe that we must reinvent the wheel; but at the same time, perhaps because of the legal perspective that I have on some of

these issues, I see opportunities to improve that process. I wish you a most successful Symposium and I thank you for you thoughtful attention.

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Introduction

The last five years have been an interesting period in the regulation and development of the Nation's high-level radioactive waste disposal system. Recently, long after Congress mandated promulgation of standards, there has been a plethora of guidance and advice from numerous committees and scientific groups. Since EPA is required to reissue the disposal provisions of our standards, it is an appropriate time for us to receive this advice. Much of the advice, however, concerns major changes in well established provisions of laws and regulations that would require in-place systems be torn down and new ones built. There certainly needs to be a careful analysis made of the present system's viability and a consensus reached before such a drastic and expensive step is taken. Personally, at least from a regulatory perspective, I am not convinced such steps are necessary.

In my remarks today I do not intend to discuss the larger societal and philosophical issues involved in disposal of high-level radioactive waste. Rather, in the first portion of this talk I will discuss some of the regulatory advice offered in recent reports and will indicate EPA's views on the issues involved. While we in EPA agree with most of the recent advice, there are at least several areas in which we disagree. I will focus my final remarks on three of these issues.

Highlights of Reports

ICRP 46-Radiation Protection Principles for the Disposal of Solid Radioactive Waste (1985)

In 1985, the ICRP issued a report that discussed how the principles of radiation protection could be applied to the problem of radioactive waste disposal. They pointed out that the principles of justification and optimization should be retained, that normal releases should be subject to annual dose limits, and that some exemptions from disposal regulations were appropriate. These are all views that we generally agree with and have incorporated in our standard setting process. However, the committee also called for probabilistic risk limits to be applied

to individual annual radiation doses. This is one of the areas of disagreement that I will address in more detail later.

NEA PAAG/DOC(90)4 Disposal of Radioactive Waste-Review of Safety Assessment Methodologies

This document, developed by the Committee on Radioactive Waste Management and still undergoing final review by the OECD/NEA, reviews the performance assessment capabilities for radioactive waste disposal. Although much of the report discusses techniques that are beyond the scope of this presentation, some of its findings are worth highlighting here.

The task group noted that "performance assessment is multi-disciplinary and iterative in approach." They also noted that "the calculated long term consequences of a repository must be considered with respect to their probability of occurrence." In this context they also stated, "However, in most cases of probability estimates, human judgement has to be used in conjunction with incomplete or only partially relevant data and observations." These are concepts that we endorse and have included in the high-level waste standards we promulgated in 1985. Like so many others,they also indicated that more work needs to be done and that "It is not obvious, however, how compliance should be demonstrated for the long term safety of repositories."

NEA RWM/DOC(90)2 RWMC Collective Opinion on Safety Assessment

The same NEA committee has followed up its review of methodologies to develop this draft collective opinion. In carrying out this effort they considered whether 1) disposal systems and their impacts on people and the environment could be sufficiently understood, 2) specialist and regulatory authorities could be convinced that the predicted behavior is representative of what might actually happen, and 3) the potential impacts and means of estimating these can be illustrated transparently for a wider audience. They concluded that "...safety assessment methodologies exist today to illustrate the long-term radiological impacts that a proposed radioactive waste disposal system could have on man and his environment." We agree with this conclusion.

GAO NUCLEAR WASTE-Quarterly Report as of December 31, 1989-(Published April 1990)

This General Accounting Office (GAO) quarterly report is particularly pertinent to today's discussion. The report reflects some NRC staff concerns as to whether the EPA containment requirements may make it difficult, if not impossible, to satisfactorily demonstrate compliance in an NRC licensing proceeding. GAO notes that "Specifically, the staff believes that the standard can be implemented successfully in a

licensing proceeding only if the inherent uncertainties involved in making long-term projections of repository performance can be satisfactorily taken into account." They further note, however, that "NRC's staff believes that meaningful, though not statistically rigorous probability estimates can be developed and reasonably defended for repository sites that are not complex or unusually geologically active. In fact, the staff believes that the required probability estimates will help determine how well a site is understood and, therefore, how much confidence can be placed in its future performance as part of a repository."

This subject of concern over uncertainties and how they are handled in an NRC licensing forum is the second of the three areas I will discuss later.

NRC Advisory Committee on Nuclear Waste letter to Chairman Carr of May 1, 1990; Subject: CRITIQUE OF THE ENVIRONMENTAL PROTECTION AGENCY'S STANDARDS FOR DISPOSAL OF HIGH-LEVEL WASTES

This NRC advisory committee has been conducting a review of the EPA High-level Radioactive Waste Standards over an extended period. They, too, indicate concern over showing compliance in the context of an NRC licensing hearing. Also, as in some of the other advisories, they state: "Although lower level standards can be stated probabilistically, they should be expressed in terms of annual risk limits from a disposal facility in an undisturbed and a disturbed state." This is, of course, an area where I have already noted I will have further comments.

This committee has also made several other specific suggestions concerning the EPA standards. We have evaluated those suggestions and have asked for clarification on several of them.

IAEA Safety Series No. 99- Safety Principles and Technical Criteria for Underground Disposal of High-Level Radioactive Wastes (1989)

This report reflects a number of the precepts that have become the basic criteria for high-level waste disposal. One of the overlying objectives noted the largely accepted approach to the intergenerational question and the role of institutional controls. The report states that the objective is "to isolate high-level wastes from the human environment over long time-scales without relying on future generations to maintain the integrity of the disposal system, or imposing upon them significant constraints due to the existence of the repository." It is because of the logic of such an objective that we have required that analytical assessments not show dependence on active institutional controls for longer than 100 years.

For limits on exposure from gradual processes, the document

recommends the application of upper-bound dose limits that are less than the ICRP recommended 100 millirems per year. This is to prevent the overall limit from being exceeded by multiple sources. This is the approach that EPA has taken.

The report recommends considering the risks of disruptive events in a probabilistic approach using individual risk based on the ICRP-46 approach. The recommended limit for these events is a health effects risk increase of one in a hundred thousand per year. This is a higher risk than the EPA usually uses. The report also differs, as have several of the others, from the EPA decision to use total releases rather than annual dose for the probabilistic criteria.

National Research Council Board on Radioactive Waste Management Position Statement "Rethinking High-Level Radioactive Waste Disposal" (1990)

This document is critical of the whole U.S. high-level radioactive waste program. It basically calls for redoing the entire system beginning with the law. Concerning regulations, the statement implies that we would be better off without so much quantitative regulation and that we should just move forward studying the matter and doing the best possible job. This is the third area I will be specifically addressing later.

This report does contain some recommendations that are in line with suggestions we have made. For instance, it calls for performance assessments to be done on an iterative basis, an area we had suggested to DOE regarding their WIPP assessment. We further agree that one should not expect to get an analysis right the first time. Indeed, one of the major purposes of early site assessments should be to ascertain the significant areas requiring further examination.

The report makes three specific recommendations for EPA's consideration:

1) We should reconsider the detailed performance standards to determine how they will affect the level of health risks that will be considered acceptable.

We are doing this as a part of our repromulgation effort. It includes a comparison of the standard with other risk management standards EPA has promulgated in the last five years.

2) We should reexamine the use of quantitative probabilistic release criteria and examine what will constitute a reasonable level of assurance.

This is an area we are reviewing, but it should be realized that this is only partly our responsibility, since it largely

falls to the NRC in their licensing process. Our standards authority is restricted to general applicability, and most of this determination is clearly related to site-specific issues. One of the issues that I will discuss later will highlight how EPA has considered this matter of "reasonable assurance" in the drafting of its standards.

3) The report notes that all other countries use only a dose requirement and that the EPA should consider doing the same.

This, again, is the area where we most consistently disagree with many of the recommendations that have been made.

The Nuclear Waste Technical Review Board- First Report to the U.S. Congress and the U.S. Secretary of Energy (March, 1990)

This is the first in what can be expected to be a series of reports from this statutorily created advisory panel. It contains many excellent suggestions on the subjects that need to be examined during the assessment of potential repositories. It points out the critical need for preliminary performance assessments to see if the computations are possible for a site and whether any characteristics that would disqualify the site have been detected.

The report contains a listing of six different comments based on a review of a preliminary draft of EPA's reproposal of 40 CFR 191. We are adding several statements to clarify the areas that the report found ambiguous. We are also giving special attention to comments that call for changes in the standard. As the report suggests, we have already decided to drop the ALARA requirement from the next draft. We are also exploring the ^{14}C release issue, and we agree that this needs to be understood.

EPA Issues

As I mentioned earlier there are then at least three areas where EPA has differences with some of these advisories. First, should there be quantitative standards before a repository is developed; second, what level of compliance assurance does EPA believe to be appropriate; and third, why have we chosen to express the probabilistic-related part of our standard in terms of total releases rather than individual annual dose? I will now explore these issues in more depth.

The Need For Quantitative Probabilistic Standards

Probabilistic standards are necessary because of the long time period over which one must judge the repository's suitability. Without taking the probability of events into

consideration, a standard has no meaning for these types of facilities. If a site has any type of geological integrity at all, and all proposals certainly indicate this will be the case, the releases from undisturbed performance are not expected to be the ones of major concern. As we extend the analysis into the thousands and tens of thousands of years, we realize releases of some kind are indeed possible, despite the geological integrity. The releases that might occur in the longer term are dependent on disturbed performance and, therefore, are not susceptible to the classic type of standard that prescribes limits on "routine releases." Nor can they be brushed aside as of no consequence. The releases of concern for any reasonably considered geology usually result from such things as human intrusion or seismic disturbances. To ignore this reality is to develop standards that have no effect on the releases of concern.

If we take the disturbances into consideration and apply only deterministic standards, we have only two choices for the possible events: we assume they either will or will not occur. If we assume they will occur, it will be difficult to find a repository that can pass the test. If we assume they will never occur, or ignore consideration of these events, we will have abandoned having a meaningful standard. If we do not state these criteria in some type of quantitative terms, we will have no yardstick for decision. This will invite litigation.

Another reason we think that a quantitative standard is necessary is that it provides a criterion against which to measure success or failure. We are very much aware of the potential contentious nature of the forthcoming repository licensing process. Without quantitative standards in place that have gone through a public review and promulgation process, each proposed site will require extensive justification, much of which will be subjective. This could result in an adversarial situation. By having an existing quantitative measure, much of the contention could be avoided since both the licensing board and any subsequent court will have a yardstick against which to judge the arguments.

The final reason for quantitative standards is that we do not believe that the disposal of high-level radioactive waste can be approached on the basis of just trying to do a good job. Even the most experimental of engineering designers must have in mind some design goal. If you are designing a bridge you need to know the weight of the vehicles that will travel on it. An engineer would not think of designing an airplane without knowing the speed and load requirements. The country started out on this approach, which resulted with a proposed repository at Lyons, Kansas, that most now agree would not have been adequate. Furthermore, we do not believe that the public is willing to accept a nonquantitative standard as adequately protective of their health and the environment. Whether it is justified or

not, the public has the perception that this is some of the nastiest stuff on the face of the planet. They want to be assured that it has extra special care in its handling and disposal. If we ever expect to have public acceptance of our radioactive waste disposal practices it would be foolish to ignore this perception. We believe that quantitative standards are one way of acknowledging the concern and reflecting that we are doing something about it. To depart from this approach at this point would be to send a signal that we are ignoring public concern, which could be disastrous.

Compliance Assurance

The determination of whether a proposed repository complies with the EPA standard will not be made through measurement, monitoring, or inspection. For these facilities, we must depend on mathematical models to carry out the long-term predictions of performance upon which the decisions will be made. Again, we realize that this will be taking place in a rather contentious setting. We are also aware that there is no way to make these types of predictions over such extended time periods and have total certainty as to their correctness. In fact, we would expect that there would be a good deal of uncertainty. We have dealt with this issue by indicating in our standard that we are looking for the analysis to show only a reasonable expectation of the standards being met. We purposefully avoided using the term "reasonable assurance" because it has been extensively used in the licensing of nuclear reactors and has acquired connotations that could complicate the waste repository decision. We have not developed any quantitative definition for "reasonable expectation" because we felt that it was both premature and that it was partly the responsibility of the NRC in its licensing process. We have indicated that we would expect that human judgement be involved and that we accept that there is no way to prove the absolute truth of the models we must rely on. This is an area that we will continue to explore as we go through the promulgation process.

Although we did not numerically define "reasonable expectation", there were other areas in the 1985 promulgation of the EPA standards in which we gave guidance on how we would handle uncertainty. For instance, in discussing how the implementing agencies might assure compliance where predictions of performance are made, we said:

> Substantial uncertainties are likely to be encountered in making these predictions. In fact, sole reliance on these numerical predictions to determine compliance may not be appropriate; the implementing agencies may choose to supplement such predictions with qualitative judgments as well.

Also, to assure that there is appropriate truncation to the probabilistic analysis, we included the following guidance:

> The Agency assumes that such performance assessments need not consider categories of events or processes that are estimated to have less than one chance in 10,000 of occurring over 10,000 years. Furthermore, the performance assessments need not evaluate in detail the releases from all events and processes estimated to have a greater likelihood of occurrence. Some of these events and processes may be omitted from the performance assessments if there is a reasonable expectation that the remaining probability distribution of cumulative releases would not be significantly changed by such omissions.

Because it was clear that uncertainties were very much a part of the analysis, we included some thoughts on how they might be handled with guidance that:

> When the uncertainties in parameters are considered in a performance assessment, the effects of the uncertainties considered can be incorporated into a single such distribution function for each disposal system considered. The Agency assumes that a disposal system can be considered to be in compliance with 191.13 if this single distribution function meets the requirements of 191.13(a).

And similarly, regarding the analysis of undisturbed performance, we said:

> When the uncertainties in undisturbed performance of a disposal system are considered, the implementing agencies need not require that a very large percentage of the range of estimated radiation exposures or radionuclide concentrations fall below limits established in 191.15 and 191.16, respectively. The Agency assumes that compliance can be determined based upon "best estimate" predictions (e.g., the mean or the median of the appropriate distribution, whichever is higher).

Also, relative to inadvertent intrusion, we wanted to appropriately limit the discussion when we included the following in our guidance:

> The Agency believes that the most productive consideration of inadvertent intrusion concerns those realistic possibilities that may be usefully mitigated

by repository design, site selection, or use of passive controls (although passive institutional controls should not be assumed to completely rule out the possibility of intrusion).

As further substantiation that EPA fully understood the uniqueness of the repository venture and the uncertainties that went with it, we provided a mechanism for calling for alternative provisions (section 191.17). In describing the purpose of this provision in the preamble to the rule, EPA wanted to go on record to provide perspective for any future reviewer, such as a licensing board or court. In this context, following are some of the examples of statements we made in the preamble:

> In developing the disposal standards, the Agency has had to make many assumptions about the characteristics of disposal systems that have not been built, about plans for disposal that are only now being formulated, and about the probable adequacy of technical information that will not be collected for many years. Thus, although the Agency believes that the disposal standards being issued today are appropriate based upon current knowledge, we cannot rule out the possibility that future information may indicate needs to modify the standards.
>
> There are several areas of uncertainty the Agency is aware of that might cause suggested modifications of the standards in the future. One of these concerns implementation of the containment requirements for mined geologic repositories. This will require collection of a great deal of data during site characterization, resolution of the inevitable uncertainties in such information, and adaptation of this information into probabilistic risk assessments. Although the Agency is currently confident that this will be successfully accomplished, such projections over thousands of years to determine compliance with an environmental regulation are unprecedented. If--after substantial experience with these analyses is acquired--disposal systems that clearly provide good isolation cannot reasonably be shown to comply with the containment requirements, the Agency will consider whether modifications to Subpart B were appropriate.

As we have proceeded in repromulgating this standard, the area of guidance for implementation is one that we have given particular attention to. This is especially true where misunderstanding or lack of clarity has been pointed out to us. It should be clear, however, from these references to the 1985 version, that we never intended "absolute proof", as some have contended.

Individual Annual Dose versus Total Release

Probably the area in which we have had the most consistent difference with the various advisories is that of probabilistic-related assessment. Although we have set individual annual exposure levels for the undisturbed performance over a 1,000 year period, we have taken the approach of setting limits on total releases over 10,000-years for the probabilistic-related standards. When we started this standard setting effort, it was our inclination to use individual dose, since that was how radiation standards had always been set. It was only after we examined what it would mean to have to comply with such a provision that we switched to our present approach. We believe that approach is much more appropriate in view of the long time periods and uncertainties involved. The easiest way to show why we came to this decision is through reference to Figure 1.

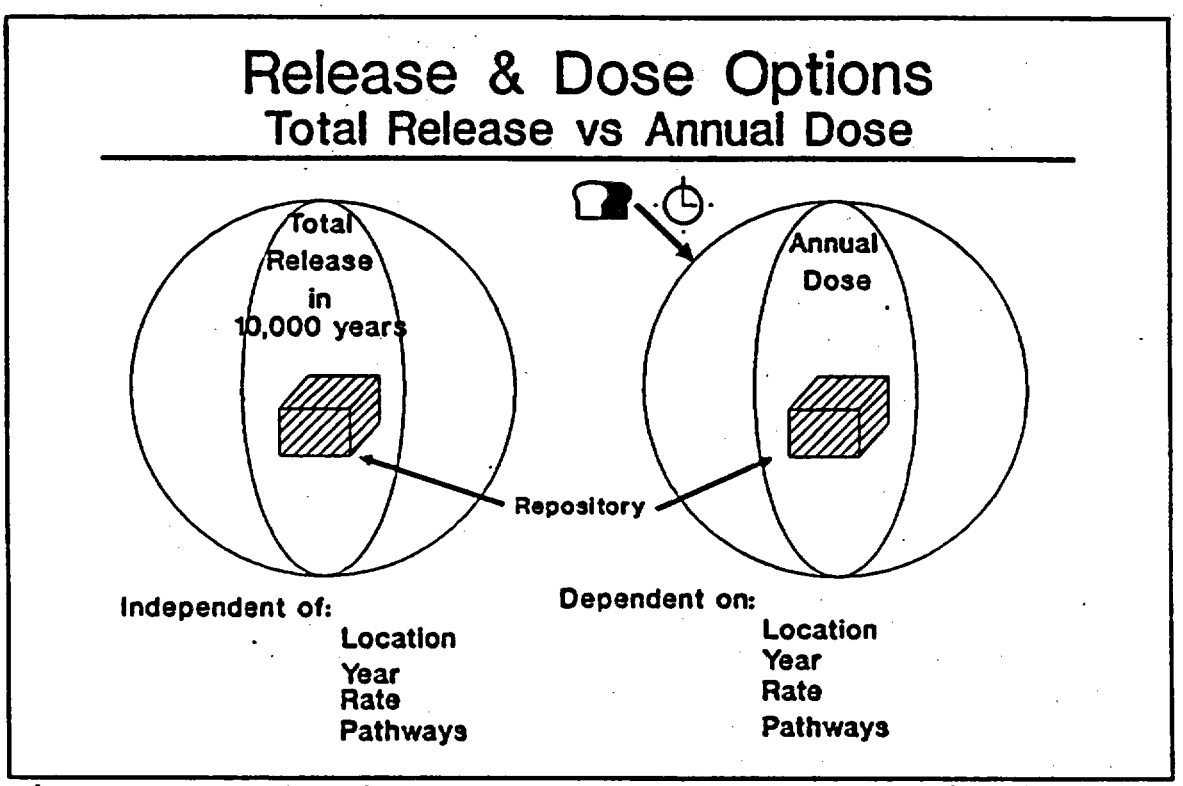

Figure 1: Total release and annual dose alternatives

Figure 1 shows two symbolic spheres representing the boundary line around two repositories and the defined "accessible environment". The sphere on the left represents the compliance case if you have the current structure of the EPA standards for probability related releases. In this instance, the release limits have been determined using fairly simple models related to the overall objective of having no greater than 1,000 health effects in 10,000 years. Under these circumstances, one only

needs to estimate the probability and quantities of releases of radioactive material across this boundary at any time during the 10,000-year period. To demonstrate compliance, it is not necessary to identify where on that boundary sphere the release occurs (Location), when the release starts or stops during that 10,000 year period (Year), the time-related frequency of the amounts released (Rate), or how this material might interact with people and how they might be exposed (Pathways).

If we were to pursue the second alternative, annual individual dose, as depicted by the sphere on the right of Figure 1, we have a much more difficult analytical task. It is no longer sufficient to just estimate how much radioactivity is likely to cross the boundary. We now must also estimate where (arrow) on that boundary it will occur and whether it is close to people. That would be only the beginning of the analytical chore, however. Since we must calculate an annual dose, we must know when the release starts and ends (clock) and how much will be released on an annual basis (time-related factors on an annual basis). While we think all this is more than should be asked of a probabilistic analysis, it would not be sufficient. We would still be required to go on from that point and speculate on how this material might interact with people (pathway-bread?) at this specific location and what the subsequent annual dose might be. These are speculations we believe are feasible for the analysis of the undisturbed repository but that are beyond what should be considered reasonable for the probabilistic analysis. Furthermore we believe that the process should be focused on the repository design and its natural and engineered barriers. With and individual probabilistic dose standard the focus could inappropriately become the pathway parameters that would be common to all alternatives and whose uncertainties might obscure the real significant choices.

CONCLUSION

There is no doubt that the country has set itself a considerable challenge in seeking to establish a high-level radioactive waste repository. Yet, we must do it. In our evaluation of the regulatory aspects of this issue we do not see the system as broken beyond repair. Certainly, the political and sociological issues appear to be much greater deterrents to success than do the technical requirements of an EPA standard. We have noted a large number of advisories concerning how these facilities should be regulated, some of which we have highlighted here. There are many ideas in these advisories that we agree with and have adopted. We find a few ideas we do not agree with, and we have pointed out our reasons for disagreeing at this meeting. We think that more such interchanges should take place and that improved communications would be helpful in resolving differences. In many ways we have an advantage because

everything we do is subject to public review, generating comments from many sectors. It would be much more difficult to create responsible public policy from discussions among ourselves. We appreciate the opportunity to have participated in this exchange.

COMMENTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE OF THE U.S. NUCLEAR REGULATORY COMMISSION

General Introduction

In June 1988, the U.S. Nuclear Regulatory Commission established the Advisory Committee on Nuclear Waste (ACNW). The Committee reports to and advises the Nuclear Regulatory Commission (NRC) on aspects of nuclear waste management within the purview of NRC's regulatory responsibilities. The focus of the Committee's work is largely on disposal but also includes other aspects such as handling, processing, transportation, storage, and safeguarding of nuclear wastes including spent fuel, nuclear wastes mixed with other hazardous substances, and uranium mill tailings. In performing its work, the Committee examines and reports on specific areas of concern referred to it by the Commission. The Committee is authorized to undertake other studies and activities on its own initiative related to those issues directed by the Commission.

In its first two years of existence, the Committee held 21 general meetings and several working group sessions and issued 37 letter reports. In addition, the Committee routinely met with the Nuclear Regulatory Commission to discuss items of mutual interest and concern.

Currently, the Committee is authorized a maximum of four members. Members are appointed by the Nuclear Regulatory Commission.

The ACNW traces its history back to the Advisory Committee on Reactor Safeguards (ACRS). The first Chairman and Vice-Chairman of the ACNW (Drs. Moeller and Steindler, respectively) had served on the ACRS where they participated extensively in the waste management reviews by the ACRS. They now continue this function with the ACNW. The current members of the ACNW are:

ACNW MEMBERSHIP

CHAIRMAN: Dr. Dade W. Moeller, Professor of Engineering in Environmental Health, School of Public Health, Harvard University, Boston, Massachusetts

VICE-CHAIRMAN: Dr. Martin J. Steindler, Director, Chemical Technology Division, Argonne National Laboratory, Argonne, Illinois

MEMBERS: Dr. William J. Hinze, Professor, Department of Earth and Atmospheric Sciences, Purdue University, West Lafayette, Indiana

Dr. Paul W. Pomeroy, President, Rondout Associates, Incorporated, Stone Ridge, New York

Today, we will be providing a summary of the advice given to the Nuclear Regulatory Commission on EPA's proposed high-level waste standards and ACNW comments on the NRC staff's review of the DOE Site Characterization Plan (SCP) for the proposed high-level waste repository at Yucca Mountain, Nevada.

EPA STANDARDS

For more than five years the ACNW and its predecessor organization have been concerned that the current set of proposed EPA standards is overly stringent, is wasteful of resources, and cannot be implemented. These concerns are based on extensive meetings and discussions with a wide range of organizations, including relevant Federal and State agencies as well as industrial and private groups. One of the highlights of these interactions was a meeting held at the Committee's conference room in Bethesda, Maryland, on March 23, 1990. The Committee continues to doubt that compliance with the EPA standards can be demonstrated for a specific repository site, even with reasonable application of the caveats included in the currently proposed standard, such as the "reasonable assurance" phrase that allows for certain flexibilities in the interpretation of probabilistic analyses. Regardless of the schemes proposed to resolve uncertainties in applying probabilistic techniques (e.g., rulemaking), the Committee has seen no convincing evidence that the current set of standards will prove to be workable.

The ACNW has concluded that the EPA standards need to be revised and that now is the time to accomplish this task. The Committee has even suggested several organizations whose recommendations for change should be sought, including the National Academy of Sciences. In such a revision, the Committee recommended that the standards should be organized in a hierarchical structure with the higher levels expressing the objectives in a qualitative sense and the lower levels stating the objectives quantitatively. The Committee stressed that the several levels be consistent and that lower levels not be more stringent or conservative than the higher levels so that they become de facto new standards. The Committee believes that the proposed quantitative EPA standards may be internally inconsistent. In addition, we believe that secondary requirements, if expressed in the EPA standards, should be given only as guidance, with qualifying statements clearly specifying that they are not to be applied in a regulatory sense.

Three principal Committee recommendations for revising the EPA standards are:

1. An acceptable risk from a high-level waste repository should be defined and justified, keeping in mind the benefits derived from the activity involved, and other societal risks as well as additional relevant considerations. Lower-level standards should be expressed in terms of annual risk limits from a disposal facility in an undisturbed and a disturbed state. The critical population group being considered should be clearly defined. This approach is in accord with recommendations of organizations such as the International Commission on Radiological Protection and the United Kingdom's National Radiological Protection Board.

2. It should be specified that inclusion in the standards of an appropriate probabilistic approach is acceptable to the definition of risk from a repository, only if it is clearly noted that this probabilistic approach is not the single determining factor in judging the acceptability of a specific site. Experience has shown that probabilistic risk analyses (PRAs) alone cannot be used to reliably determine the compliance of a single nuclear power plant with a set of standards or as the basis for judging the adequacy of its safety. A single high-level waste repository, which is to function for thousands of years, is still more difficult to assess quantitatively. The EPA standards should clearly specify that risk assessments are but one of several tools for the evaluation of a given high-level waste repository site and/or facility and that PRAs should be only one factor in evaluating compliance of such a facility with the EPA standards. Expert opinion and deterministic criteria are of considerable importance in judging the acceptability of a specific site.

3. Evaluations of the anticipated performance of the proposed Waste Isolation Pilot Plant indicate that, for the disturbed state, human intrusion is the dominant contributor to risk. Early indications suggested that performance analyses for the proposed Yucca Mountain repository may also show human intrusion to be important. For these reasons, separate considerations for evaluating the impacts of human intrusion should be included. The Committee suggested that the standards be rewritten to separate the evaluation of anticipated repository performance into three parts: (a) the undisturbed repository; (b) the disturbed repository, exclusive of human intrusion; and (c) the repository as it might be affected by human intrusion. This would clearly separate out the issues surrounding human intrusion and permit it to be addressed directly.

Currently, the NRC staff and the ACNW are moving toward, but are not yet at, a consensus over how the EPA standards must be revised. The ACNW will continue its evaluation of the EPA standards.

ACNW Review of the NRC Analysis of the DOE Site Characterization Plan

The stringency of the EPA standards, coupled with their probabilistic base, has led to the need for extensive plans for conducting studies and for collecting the data necessary for the analyses associated with determining whether a given waste disposal site can be demonstrated to show compliance. As a result, the ACNW has devoted considerable time and effort in reviewing the DOE SCP and the NRC staff's review of this plan, the Site Characterization Analysis (SCA). The ACNW review of these documents was, of necessity, less than comprehensive. Rather, the Committee focused on specific critical topics. Members and consultants reviewed relevant material in-depth, using an iterative process with the assistance of the NRC and DOE staffs. The Committee was in general agreement with the overall content of the SCA. However, the Committee had several significant concerns, some of which are summarized below:

- Statements are absent in the SCP addressing the systematic and early identification and evaluation of potentially disqualifying features at the Yucca Mountain Site. Although the SCP is an action plan for site characterization, the Committee believes a much stronger focus should be placed on early detection of potentially disqualifying features. The Committee concluded that the SCA should point to the need in DOE's SCP for an integrated section of the plan that explicitly addresses the activities leading to an evaluation of the characteristics of the site directly related to disqualifying features (e.g., groundwater travel time as stated in the NRC regulations).

- Insufficient attention is given in the SCP to the limitations and uncertainties in the Yucca Mountain data bases, and the associated difficulties in demonstrating that the repository will comply with EPA's high-level waste standards (40 CFR Part 191). Here, the key factor is that the standards, as currently written, are probabilistic and therefore the methods for demonstrating compliance must have a probabilistic base. The approach required to be used includes the construction of a complementary cumulative distribution function (CCDF) and, through this process, a demonstration that the repository complies with the EPA standards. Primary concerns of the ACNW are the uncertainties and limitations in the data to be used to construct the CCDF.

Since the ability to resolve these uncertainties experimentally may well be beyond the capability of the site characterization program, increased consideration should be given to the feasibility of developing deterministic criteria for judging the adequacy of the site relative to the EPA goals. As stated previously, the Committee considers the demonstration of compliance of the proposed repository with the EPA standards to be a major concern.

- The ACNW raised its concern over the delays by DOE in implementing satisfactory quality assurance (QA) programs. The Committee urged that this troublesome issue be resolved promptly, since continued absence of approvable QA systems will increase the burden on the participants in the licensing processes when qualification of data is at issue.

In addition to the above, the Committee offered a number of comments pertaining to other specific aspects of the site characterization program, such as resolving the dilemma of how to determine the characteristics of the Calico Hills Formation, while still maintaining this structure as a barrier between radioactive wastes placed in the repository and the underlying saturated zone, and the need to define the materials to be used in the waste packages and the manner in which these packages will be sealed. The latter information is essential to the evaluation of possible interactions between the waste package and repository materials.

We appreciate the opportunity to participate in today's discussion and look forward to an interesting exchange of information. The success of the nation's nuclear energy program will be measured in part by the skill used to manage nuclear waste. This task clearly requires the participation of people who are expert in a wide range of fields. This meeting is an important contribution to the process and thereby to the quality of the product.

Section 2 - Basis for the Standard

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PROPOSED EXTENSIONS OF UNITED STATES FUNDAMENTAL AND DERIVED STANDARDS FOR HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTE DISPOSAL

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ABSTRACT

The development of the present United States standards for transuranic and high-level waste fundamental criteria, derived release limits, and risk limits for probabilistic releases is traced through supporting documentation. The development procedures and the resulting regulations are compared to requirements for the standards, traditional methods of regulating chemical and radiological carcinogens, and recommendations made by the International Commission on Radiological Protection, the Nuclear Energy Agency, the Board on Radiological Waste Management, the Science Advisory Board, the Nuclear Waste Technical Review Board, and individual investigators. The development methodology, logic, assumptions, and models are reviewed relative to the two proposed repositories. Individual difficulties are defined, and their probable causes and potential effects are examined. Several options are suggested for modifying and extending the standards for each of the four major areas; these options make the standards more appropriate for the sites now under consideration, relate them more directly to the actual safety of the repositories, and make them more defensible. Many of the extensions are compatible with the present standards and would not require any significant changes in philosophy, methodology, or format. The benefits of an enhanced quality assurance program are discussed and several other changes in development procedures for waste disposal regulations are suggested.

UCB-NE-4006
August 1981

Derivation of EPA Proposed Standard for Geologic Isolation of High-Level Waste

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The EPA proposed standard, presented in Table 1, lists cumulative release limits of individual radionuclides to the accessible environments for a period of 10,000 yr after disposal. The present addendum summarizes my understanding of how that standard was derived.

The release limits were not derived from EPA's calculations of release from conceptual repositories. Instead, EPA evidently derives the release limits entirely from equations equivalent to those on page 6 of UCB-NE-4006. EPA assumes groundwater from a repository discharges into a river. They then calculate the radiation dose and health effects to man from several different pathways, including drinking the contaminated river water, irrigation of crop-producing land and consumption of contaminated produce, resuspension and inhalation of radioactive material from land surfaces, etc. The EPA approach towards calculating the population dose from drinking river water is illustrated here. The drinking water pathway is denoted as pathway $t = 1$. They calculate the quantity V_T of fresh water produced in the world per year and the quantity V_1 of fresh water ingested by an individual per year. For a total world population of P people, the fraction f_1 of fresh water ingested by people is

$$f_1 = \frac{PV_1}{V_T} \qquad (1)$$

EPA then estimates the curie quantity Q_i of radionuclide i released in 10^4 years from a 10^5 Mg U-equivalent repository. They assume that the curies ingested is given by

$$\text{Ci of i ingested} = Q_i f_1 f_2 \quad (2)$$

where f_2 is the fraction of released radionuclides which enter fresh water supplies.

Multiplying by the number P_{i1} of premature cancers in the population drinking the river water, per curie ingested by the population, EPA calculates the number of premature cancers N_{i1} from ingestion:

$$N_{i1} = Q_i f_1 f_2 p_{i1} \quad (3)$$

Contrary to what is said in UCB-NE-4006, EPA does not obtain the risk per curie from the ORNL code INREM-2. Instead, EPA has made its own calculations of the risk per curie, with results given in the draft report by Smith, Fowler, and Goldin[2].

To estimate f, EPA assumes:

$p = 10^{10}$ people, assumed to be constant for 10^4 yr. Present world population is 3.8×10^9 people.

$V_T = 3 \times 10^{16}$ ℓ/yr

$V_1 = 603$ ℓ/yr per individual

These values in Eq. 1 result in:

$f_1 = 2 \times 10^{-4}$

No value of f_2 is given, but evidently EPA assumes this to equal unity. EPA data for the product $f_1 p_i$ are evidently given in the third column of Table D-2, page 149, of the report by Smith, Fowler, and Goldin[2].

With a similar approach, EPA estimates the number of premature cancers $N_{i\ell}/Q_i$ per curie released for pathway ℓ, for each of the 30 pathways considered in their analysis. The total number of premature cancers per curie released is then obtained by

$$N_i/Q_i = \sum_{\ell=1}^{30} \frac{N_{i1}}{Q_i} \qquad (4)$$

EPA data[2] for N_i/Q_i are listed in the second column of Table 1.

To calculate a release limit Q_i for radionuclide i, EPA specifies a maximum of 1000 premature cancer deaths in 10,000 yr from a repository containing waste equivalent to 10^5t of uranium fuel*. They then renormalize to 10 premature cancers over 10,000 yr per 1,000 t of uranium fuel. Setting N = 10 premature cancers, if radionuclide i is the only source of radiation dose, the normalized release limit Q_i is obtained by

$$Q_i = \frac{10}{N_i/Q_i} = \frac{10}{\sum_{\ell=1}^{30} \frac{N_{i1}}{Q_i}} \quad \text{curies/1,000 t} \qquad (5)$$

Release limits for selected nuclides, calculated from Eq. (5), are shown in Table 1. Also shown are the release limits from the EPA proposed standard. Evidently EPA rounds off its calculated limits to one significant figure. The calculated limits agree with those in the EPA proposed standard for all nuclides except for ^{99}Tc, where the calculated value is seventeen-fold greater. The basis for the release limit for ^{226}Ra is not given.

*A 10^5 -t repository will contain the waste from 100 1000-MWe nuclear plants, each operating for its plant life of about 40 yr. The EPA safety goal of N_i = 1,000 is equivalent to 10 premature cancer deaths over 10^4 yr from the high-level waste from a single 1000 MWe plant.

Table 1. Calculated release limits and release limits in EPA proposed standard.

Nuclide	Health effects per curie released to a river a/	Release limit per 1000 t of uranium fuel, Ci in 10^4 yr.	
		From Eq.(4)	From EPA proposed standard
^{14}C	4.58×10^{-2}	2.18×10^{2}	2×10^{2}
^{90}Sr	1.21×10^{-1}	8.26×10^{1}	8×10^{1}
^{99}Tc	2.85×10^{-4}	3.5×10^{4}	2×10^{3}
^{126}Sn	1.20×10^{-1}	8.33×10^{1}	8×10^{1}
^{129}I	1.08×10^{-2}	9.26×10^{2}	9×10^{2}
^{135}Cs	3.81×10^{-3}	2.62×10^{3}	2×10^{3}
^{137}Cs	1.98×10^{-2}	5.05×10^{2}	5×10^{2}
^{226}Ra			3
^{237}Np	5.96×10^{-1}	1.68×10^{1}	2×10^{1}
^{238}Pu	2.29×10^{-2}	4.37×10^{2}	4×10^{2}
^{239}Pu	6.92×10^{-2}	1.44×10^{2}	1×10^{2}
^{240}Pu	6.53×10^{-2}	1.53×10^{2}	1×10^{2}
^{242}Pu	6.76×10^{-2}	1.48×10^{2}	1×10^{2}
^{241}Am	7.19×10^{-1}	1.39×10^{1}	1×10^{1}
^{243}Am	2.68	3.73	4

a/ $\sum_{l=1}^{30} \frac{N_{il}}{Q_i}$ from Smith, Fowler, Goldin[2], p. 148.

References:

1. T. H. Pigford, "EPA Proposed Standard for Geologic Isolation of High-Level Waste", UCB-NE-4006, March, 1981.

2. J. M. Smith, T. W. Fowler, A. S. Goldin, "Environmental Pathway Models for Estimating Population Health Effects From Disposal of High-Level Radioactive Waste in Geologic Repositories", EPA 520/5-80-002 (June 8, 1981) Draft.

GASEOUS RELEASE OF CARBON-14: WHY THE HIGH LEVEL WASTE REGULATIONS SHOULD BE CHANGED

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ABSTRACT

The high-level nuclear waste regulations pertaining to gaseous release of carbon-14 from a repository should be changed to allow greater release, for several reasons. Some of them are as follows. First, the total amount of carbon-14 that would be placed in a repository is small compared to that produced naturally in the atmosphere by cosmic rays. Second, the dose that would result to an individual from total release of repository carbon-14 would be very small compared to that from natural radiation sources and would be well below the "Below Regulatory Concern" criterion. Third, the limits on gaseous carbon-14 release from a repository have been set unreasonably low compared to the limits set for carbon-14 release from other fuel cycle facilities. Fourth, the additional cost for waste packages to attempt to meet the regulations for carbon-14 release would likely be of the order of a billion dollars or more, too high to be justified by the small reduction in dose that might result.

INTRODUCTION

The issue of carbon-14 release from spent fuel waste packages has been of concern to those involved in waste package design and performance analysis for the Yucca Mountain Site Characterization Project (formerly known as the Yucca Mountain Project, formerly known as the Nevada Nuclear Waste Storage Investigations Project) since 1983. The present author coauthored two papers on the subject in 1984[1] and 1986,[2] respectively. At the request of the Office of Civilian Radioactive Waste Management, the author wrote a draft review and position paper on this subject, which was circulated for comment within the waste management community beginning in September, 1989. Five papers[3-7] were presented on carbon-14 by other authors at the first annual International Topical Meeting on High Level Radioactive Waste Management in April 1990. The Nuclear Waste Technical Review Board, the Board on Radioactive Waste Management of the National Research Council, and the Advisory Committee on Nuclear Waste have each received briefings on this subject. Analyses of this issue have been carried out by the Nuclear Regulatory Commission staff and by Arthur D. Little, Inc. for the Environmental Protection Agency.

Carbon-14 is present only in small quantities in spent reactor fuel. However, it is readily oxidized to $^{14}CO_2$ in the presence of oxygen at elevated temperatures, and can then be transported in the gaseous phase. This is particularly relevant to the possibility of locating a repository in Yucca Mountain, should the site be found suitable, since oxygen would be present, and gaseous transport could occur in the unsaturated zone under consideration.

The limits on release of carbon-14 have been established by the Environmental Protection Agency under 40CFR191 (currently under reconsideration by EPA) and by the Nuclear Regulatory Commission, under 10CFR60. As a result of these regulations, release of carbon-14 is at present the dominant factor determining the required containment lifetimes and the allowable container failure rates for spent fuel packages for a repository, if constructed at the Yucca Mountain site.

GENERAL INFORMATION ABOUT CARBON-14

Carbon-14 is produced naturally by interactior of cosmic rays with gases in the upper atmosphere, resulting in spallation neutrons, which in turn undergo capture reactions. The main ^{14}C production reaction is $^{14}N(n,p)^{14}C$. The estimated rate of production is 28 kilocuries per year.[8] The main chemical form of ^{14}C in the atmosphere is $^{14}CO_2$. Carbon-14 has a half-life of 5,730 years. It decays by beta emission with a maximum beta energy of 0.155 MeV.[9] It also exchanges with other terrestrial carbon reservoirs on various timescales: surface mixed layer of the oceans — a few years; deep oceans — hundreds of years; marine and terrestrial sediments — thousands of years.[10] The

resulting balance between production, exchange, and decay led to a more or less steady atmospheric concentration before the 1800's of 7.5 ± 2.7 pCi per gram of carbon,[11] which is the basis for radiocarbon dating.

MANKIND'S INTERACTION WITH ATMOSPHERIC CARBON-14

The industrial revolution introduced large-scale combustion of fossil fuels, which are too old to retain carbon-14. This decreased the atmospheric ^{14}C specific activity by about 2% by the 1950's, a phenomenon known as the Suess effect.

Atmospheric nuclear weapons tests in the 1950s and 1960s nearly doubled the atmospheric ^{14}C specific activity.[12] It has since decreased by exchange processes to about 15% above the natural value and is continuing to decrease.

The atmospheric inventory was estimated in 1977 to be 3.8 megacuries.[8] The global inventory was estimated at 230 megacuries,[8] mostly as bicarbonate in the oceans. Carbon-14 enters the biosphere primarily by photosynthesis of plants, followed by ingestion of plants by animals. Equilibration of the ^{14}C specific activity of the human body with that of the atmosphere occurs with a lag time of only about 1.4 years.[13] Thus the radiation dose rate to the individual from carbon-14, which results from that incorporated in his own body, is directly proportional to the specific activity of ^{14}C in the atmosphere.

CARBON-14 PRODUCTION IN LIGHT WATER REACTORS

Carbon-14 is produced in the fuel assemblies (UO_2, cladding, and fuel assembly hardware), non-fuel assembly structural materials, and reactor cooling water of light water reactors.[14] The main production reaction in the fuel, cladding, and structural metals is $^{14}N(n,p)^{14}C$, in which thermal neutrons are captured by impurity nitrogen. The main reaction in the water is $^{17}O(n,\alpha)^{14}C$. Most of the ^{14}C produced in the cooling water is exhausted to the atmosphere, at rates of about 10 curies per year for a 1000-MWe plant.[15] A small fraction goes into ion exchange resins and is disposed of as low-level waste.[16] The ^{14}C produced in the fuel assemblies is mostly retained in the spent fuel, although some is lost from the external surface by dissolution in the water of spent fuel pools, followed by exchange with ventilation air and exhausting from the stack to the atmosphere. Reactor structural material not associated with fuel may end up as low level waste or as "greater than class C" waste at the time of replacement or decommissioning.

INVENTORY OF CARBON-14 IN SPENT FUEL

The ^{14}C inventory in spent fuel can be estimated by calculations and measurements. Calculations require a knowledge of the average nitrogen impurity content of UO_2, zircaloy, and structural metals in the fuel assemblies. The most comprehensive calculation for U.S. fuel was done by Davis at ORNL in 1977.[14] Subsequent work by ORNL[17-19] has used these values for nitrogen content: UO_2 — 25 μg/gU, zircaloy - 80 wppm, 304 stainless steel — 1300 wppm, and nickel alloys — 1300 wppm.

I have re-examined these nitrogen concentrations. The UO_2 value appears to be well-founded, since it was based on measured data at reactor fuel manufacturing plants. The other numbers appear to be conservatively high, since they were based on maximum values from consensus standards. I have therefore chosen new values which more nearly reflect actual production averages. For zircaloy I have adopted the value 40 wppm, based on discussions with John Schemel of Sandvik Special Metals, Richland, WA in 1989. For 304 stainless steel, I have adopted the value 400 wppm.[20] For the nickel alloys I have adopted the value 120 wppm, based on discussions with James Crum of INCO Alloys International of Huntington, West Virginia in 1989. This reflects current use of vacuum melting, or degassing by means of an argon-oxygen blowing process.

To calculate the revised carbon-14 inventory in spent fuel, I scaled the ORNL results for each component by the ratio of the new nitrogen concentrations to the old ones. In doing so, I also used new measurements of the variation of neutron flux and spectrum-averaged cross sections with height in reactor cores.[21]

As an example, my revised result for the ^{14}C inventory of PWR spent fuel with a burnup of 33,000 MWd/MTU is 1.00 curie per metric ton of initial uranium, as compared to the earlier ORNL value of 1.55. For BWR spent fuel with a burnup of 27,500 MWd/MTU, my new value is 1.02 curies per metric ton, compared to the ORNL value of 1.53. This includes the zircaloy fuel channel. Comparison to measurements made by the Materials Characterization Center at Pacific Northwest Laboratories has been made for two fuel assemblies for which all the components have been analyzed for ^{14}C (data received from R.J. Guenther and J.E. Mendel in 1989). In one case, the measured ^{14}C inventory was 25% greater than my revised result. In the other case, it was 27% lower. In order to obtain more precise average values, a large number of fuel assemblies would have to be analyzed. Since a large proportion of the fuel destined for the repository

has not yet been manufactured, it appears that these revised calculated values are the best values we will have to work with for the foreseeable future.

ESTIMATE OF REPOSITORY CARBON-14 INVENTORY

The following basis was used along with the above-described revised carbon-14 inventory for spent fuel: The total repository capacity is 70,000 metric tons of initial uranium equivalent, of which 7,000 is assumed to be defense waste and 63,000 is spent fuel (Defense waste does not contain ^{14}C, since it was exhausted to the atmosphere during reprocessing). The average burn-ups are taken to be 29,500 MWd/MTU for BWR fuel and 37,500 for PWR fuel. On a tonnage basis, the repository is assumed to receive 35.7% BWR spent fuel and 64.3% PWR spent fuel.

The results are an average of 1.12 curies per metric ton uranium in the spent fuel, and a repository total of 71 kilocuries of ^{14}C.

PHYSICAL DISTRIBUTION AND CHEMICAL FORMS OF CARBON-14 IN SPENT FUEL

As discussed above, ^{14}C is produced in the UO_2, the zircaloy, and the fuel assembly hardware. For PWR spent fuel with a burnup of 33,000 MWd/MTU, the calculated ^{14}C distribution for these three components is 0.60, 0.18, and 0.22 curies/MTU, respectively, for a total of 1.00 curies/MTU. For BWR spent fuel with a burnup of 27,500 MWd/MTU, the corresponding values are 0.54, 0.38, and 0.10 curies/MTU, respectively, for a total of 1.02 curies/MTU. There is a negligible amount of ^{14}C in the fuel rod gas.

The chemical forms of ^{14}C in spent fuel are not known, but may include elemental carbon, carbides, and oxycarbides.

RELEASE OF CARBON-14 FROM SPENT FUEL

Measurements performed by LLNL in 1983[1] revealed that ^{14}C was rapidly released from the outside surfaces of intact spent fuel cladding when heated in air, appearing as $^{14}CO_2$ in the air.

Subsequent work by Smith and Baldwin[22] showed that as much as 2% of the spent fuel ^{14}C inventory was released from zircaloy cladding in 8 hours at 350°C in air. The time dependence of the release was consistent with diffusion in a layer of finite thickness, suggesting that the ^{14}C is released from the oxide layer on the zircaloy. The identity of the diffusing species is not known. The temperature dependence was Arrhenius, with an activation energy between 19 and 25 kcal/mole, depending on oxide thickness. The amount of ^{14}C released in 8 hours at 200°C was about a factor of 25 lower than at 350°C. At 100°C, it was about a factor of 100 lower than at 350°C. In commercial grade argon (having about 50 ppm oxygen) the release was about a factor of 10 lower than in air. The effects of ionizing radiation on oxidation of ^{14}C in spent fuel have not yet been quantified.

Spent fuel dissolution experiments by Wilson[23] showed that an additional amount of about 1/2% of the overall ^{14}C inventory is readily accessible at the fuel-cladding gap after cladding breach. Release of the remaining inventory requires oxidation of UO_2 and metal parts.

RELEASE OF $^{14}CO_2$ FROM WASTE PACKAGES

No release of $^{14}CO_2$ can occur from waste packages unless the waste containers are breached. Similarly, no release of ^{14}C from UO_2 or the fuel-cladding gaps can occur unless the cladding is breached (a small fraction of the fuel rods are known to have initially-breached cladding).

The material from which the containers will be fabricated has not yet been selected. Consequently, the dominant container failure mode or modes are not known at present. Once the dominant modes are known, it will still be a difficult task to predict the distributions of time-to-failure and the number and size distributions of container perforations. Work is in progress to predict the distribution of time-to-failure for the cladding, and this is also a difficult task.

Researchers from LBL[4] and BNL[5] have theoretically analyzed the release of $^{14}CO_2$ through perforations of various sizes in waste containers, but it is not possible to apply their results deterministically in the absence of a knowledge of the number, size, and time distributions of perforations.

Breach of one container in 20,000 in a single year during the containment period, coupled with a 2% loss from the breached containers, would exceed the DOE interpretation of "substantially complete containment." Breach of one container in 2,000 in a single year during the controlled release period, coupled with a 2% loss from these containers, would violate the NRC 10CFR60 limit of one part in 10^5 per year.

TRANSPORT OF CARBON-14 THROUGH THE GEOLOGICAL MEDIA TO THE ACCESSIBLE ENVIRONMENT

Natural transport of gases through Yucca Mountain has been studied by the USGS.[24] Theoretical modeling of the gaseous transport of ^{14}C from a repository in Yucca Mountain has been performed by several researchers.[25-27] There appears to be a consensus that the travel time of $^{14}CO_2$ to the surface will be comparable to or less than one half-life (5,730 years). Consequently, relatively little credit can be taken for nuclear decay during transport.

ASSESSMENT OF LIKELIHOOD THAT DOE, NRC, AND EPA LIMITS ON CARBON-14 RELEASE CAN BE MET

The reference design waste package described in the site characterization plan incorporated a single-walled metal container with a wall thickness of about 1 cm. Depending on the metal selected, the cost estimates per container for this design ranged from $45,000 to $83,000 (estimates compiled by Babcock and Wilcox, to be published). Taking into account the number of packages in the repository, the cost for these containers would be in the 1 to 3 billion dollar range. This design was intended to prevent aqueous dissolution of radionuclides during the 300-1,000 year containment period. Gaseous release of ^{14}C is more difficult to prevent.

In my opinion it is unlikely that waste containers of reasonable cost could meet either the DOE interpretation of "substantially complete containment" (10^{-6} of the inventory per year) or the NRC 10CFR60 release rate limit (10^{-5} per year) for carbon-14. This opinion is based on the difficulty of assuring that initial flaws are not present in the waste containers, and on my estimate of what is possible in terms of prediction of corrosion processes over long time periods, particularly in view of inherent uncertainties in predicting a repository environment far into the future.

In my opinion, it is also not possible to give assurance that the EPA 40CFR191 cumulative release limit of 0.1 curie of ^{14}C per MTU over 10,000 years could be met. This limit would constitute only 10% of the ^{14}C inventory. When one combines the uncertainties in predicting the containment lifetimes of waste packages, the oxidation rates of the metal parts of fuel assemblies, the containment lifetimes of fuel cladding, and the oxidation rate of the UO_2 fuel over such a time span, with the rapid transport rate of $^{14}CO_2$ through the geologic media, it is not possible to state confidently that this cumulative release limit could be met. Efforts to achieve some degree of assurance would likely involve devising waste packages with gettering material and multiple walls, which would likely cost a multiple of the cost of the reference design. That is, the cost increase would be measured in billions of dollars. Prudent use of the nation's resources demands that we ask what would be gained by such an expenditure. This analysis follows.

CONSEQUENCES OF RELEASE OF THE TOTAL REPOSITORY CARBON-14 INVENTORY

As discussed above, the estimated global inventory of naturally-produced ^{14}C is 230 million curies. In comparison, the estimated repository ^{14}C inventory of 71 kilocuries is less than 1/3000 of the global inventory of ^{14}C. Taking into account the natural cosmic ray production rate of 28 kilocuries per year, the entire repository inventory would be produced naturally in less than three years.

If the entire repository inventory of ^{14}C were released instantaneously, the average atmospheric concentration would increase less than 2%. If the release occurred over a time longer than a few hundred years, exchange processes would limit the increase in atmospheric concentration to less than 0.1%. By comparison, the atmospheric nuclear weapons tests had raised the concentration in the northern hemisphere by 84% by 1964, when the atmospheric test ban treaty took effect.

The burning of fossil fuels and the manufacture of cement (roasting of carbonate minerals) release $^{12}CO_2$ to the atmosphere. This dilutes the $^{14}CO_2$, lowering its specific activity. Modeling indicates that the specific activity of ^{14}C in the atmosphere will not significantly increase as a result of world-wide nuclear power production for the foreseeable future (to the year 2050).[28] The repository inventory would constitute only a small part of the world-wide total.

Average dose rates to individuals in the U.S. and Canada from natural background radiation have been estimated at 300 mrem per year by the NCRP.[29-30] Of this total, 2/3 is contributed by radon, and 1 mrem is due to natural ^{14}C. Using the data presented above, release of all repository ^{14}C would raise this dose rate by less than 20 microrem per year, and probably less than 1 microrem per year. It is interesting to note that the NRC has established a policy that dose rates less than 1 to 10 millirem (or 1,000 to 10,000 microrem) are "Below Regulatory Concern."[31]

Estimating the dose to the maximally exposed individual depends sensitively on several

assumptions: the location where the individual's food is grown, the assumed inventory of ^{14}C, the assumed rate of release, the assumed travel time through the geologic media, the surface area over which the ^{14}C is discharged, and the speed and direction of winds. The location where the food is grown is crucial, because by far most carbon incorporated in the human body comes from ingestion rather than inhalation. Two factors will tend to keep the dose to the maximally exposed individual close to that for the global average individual. First, Yucca Mountain is located in a desert. Large scale agriculture in its immediate vicinity is unlikely. Second, the time required for dispersal by winds and atmospheric mixing is very short compared to the probable timescale of release and compared to the half-life for decay. Daer (see Ref. 7) assumed that an individual lived over the repository "footprint," and that his food was grown elsewhere. He assumed a release of 1,000 curies of ^{14}C at the surface each year, over an area of 1500 acres. The winds were based on data from Yucca Mountain. Daer's result for dose rate to the maximally exposed individual was 0.05 mrem per year. This can be compared to the 300 millirem per year natural dose rate and the 1 to 10 millirem per year "Below Regulatory Concern" limit.

It is abundantly clear from these comparisons that dose to individuals from total release of repository ^{14}C is too low to be of concern, either to global average individuals or to those living in the vicinity of Yucca Mountain. However, the EPA 40CFR191 Table 1 cumulative release limit for ^{14}C was based on integrating over a projected world population out to 10,000 years, assuming a linear, no threshold model for dose response, and limiting the health effects to 0.1 per year for 100,000 metric tons equivalent of spent fuel and high level nuclear waste. This is called the collective effective dose equivalent commitment (CEDEC) approach.

A more recent analysis of this type by McCartney et al.[28] was based on the assumptions that the world population levels out at 10 billion, and the total recoverable reserves of fossil fuels are $4x10^{15}$ kg (carbon). They used a 25-box carbon cycle model. For an integration time of 100,000 years, they obtained a CEDEC of 470 person-rem per curie of ^{14}C released. This can be compared with the EPA value of 399 person-rem for the shorter integration period of 10,000 years.[32] Using the McCartney results, if the entire repository inventory were released to the atmosphere, the total CEDEC would be 33 million person-rem. For comparison, a natural background dose rate of 300 mrem per year would produce a CEDEC of $3x10^{13}$ person-rem over 10,000 years or $3x10^{14}$ person-rem over 100,000 years. These CEDECs are a factor of one to ten million higher than that for the repository ^{14}C. Again, the release of repository ^{14}C pales into insignificance by comparison.

NONUNIFORMITY IN THE REGULATORY TREATMENT OF CARBON-14 RELEASE

Regulatory treatment of ^{14}C release is not uniform. Regulations currently allow direct airborne release of ^{14}C from reactors, and it amounts to one-quarter to one-third as much total ^{14}C radioactivity as will be present in the repository. Shallow trench burial of significant amounts of ^{14}C in the form of low level wastes is also allowed and is currently practiced. All gaseous radionuclide releases other than those from a high level waste repository are regulated under 40CFR61. Park and Pflum[7] have recommended that repository ^{14}C emissions either be regulated under 40CFR61 or that ^{14}C should be exempt from all EPA regulations. I support this view.

SUMMARY AND CONCLUSIONS

1. The repository would have a carbon-14 inventory of about 71 kilocuries.
2. About 2% of the carbon-14 is located on the outside of the fuel cladding. About 1/2% is readily accessible at the fuel-cladding gap.
3. Carbon-14 is readily oxidizable to $^{14}CO_2$ and is rapidly transported in the gas phase through the unsaturated zone.
4. For waste containers of reasonable cost, it does not appear likely that either the DOE 10^{-6} per year release rate criterion, the NRC 10^{-5} per year release rate limit, or the EPA 0.1 curie per metric ton cumulative release limit over 10,000 years can be met for carbon-14.
5. The consequences of total repository carbon-14 inventory release on human health and the environment would be extremely small in comparison to the effects of natural radiation.
6. The stringent limits placed on carbon-14 release from the repository are in sharp contrast to the higher limits applied to carbon-14 release from other fuel cycle facilities, such as operating power reactors.
7. In view of the above, it seems reasonable to conclude that the high level nuclear waste regulations should be changed to allow higher releases of carbon-14 from the repository.

8. Reasonable approaches appear to be to regulate gaseous release of carbon-14 from the repository on the same basis as all other gaseous radionuclide releases to the atmosphere or to exempt carbon-14 releases from all EPA regulations.

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EFFECT OF A LOW-PERMEABILITY LAYER ON CALCULATED GAS FLOW AT YUCCA MOUNTAIN*

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INTRODUCTION

Yucca Mountain is being studied to determine its suitability as a location for a high-level nuclear waste repository. The mountain is a steep-sided linear ridge which is underlain by a 500-meter thick unsaturated zone composed of alternating layers of ash-flow and bedded tuffs.[1]

Seasonal flows of air with velocities as high as 3.5 m/s have been observed[2] in deep boreholes at Yucca Mountain. These flows are attributed to convective circulation arising from topographic relief, seasonal temperature variation, and density differences resulting from variation in gas composition. If a repository were built there, heat from the emplaced waste would also contribute to gas flow. Large-scale air flows may be significant to repository performance because they control the movement of carbon dioxide within the mountain and also result in a net flux of water vapor to the surrounding atmosphere. An understanding of the velocity, trajectories, and mixing of the gas in Yucca Mountain is necessary both as input for a model of the carbon-14 movement in the unsaturated zone[3] and evaluation of the net vapor flux.

Amter and Ross[4] developed a model called TGIF (Topographic Induced Flow) to simulate gas flow under Yucca Mountain. The TGIF model differs significantly from previous gas flow models. It uses a governing equation that is based on the concept of freshwater head, thus avoiding the numerical problems associated with the near-cancellation of the forces due to gravity and the pressure gradient. Unlike most other models, dipping, layered media can be simulated.

This paper describes a systematic sensitivity study that was designed to test several aspects of the TGIF model when used to simulate gas flow under Yucca Mountain. Values of three important inputs to the model were systematically varied to form a matrix of 80 runs. The matrix consisted of five values of permeability contrast between a bedded tuff layer and surrounding welded units (in all cases, bulk permeabilities were used to represent the combined effect of both fractures and matrix permeability), four temperature profiles representing different stages of repository cooldown, and four finite-difference grids.

THE MODEL

The derivation of the governing equation on which TGIF is based involves the following assumptions about the physical system in the deep subsurface at Yucca Mountain:

- Thermodynamic equilibrium exists among air, water vapor, and water.
- The gas behaves as an ideal gas.
- The gas is saturated with water vapor.
- The gas flow field is at steady state.
- Changes in partial pressure of water vapor are accommodated by changes in gas composition, with the total pressure remaining nearly constant.[5]
- Gas viscosity is independent of pressure.
- Acceleration of gravity is uniform.
- Molecular diffusion resulting from gradients of water vapor partial pressure has a negligible effect on gas flow.[4]
- All gas-filled voids in the matrix may be treated as a single porosity on time scales of years (see data of Montazer et al.[6]).

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- Gas permeability is independent of pressure.
- The unsaturated zone stays at constant saturation, with water lost to evaporation replenished by precipitation or from the water table.

The validity of these assumptions at sufficiently low temperatures is well established. They imply that the system can be described by three equations, a volume balance, a constitutive relation, and Darcy's Law, as follow:

$$\nabla\cdot q - q\cdot[(\frac{1}{T}+\frac{1}{P_a}\frac{dP_v}{dT})-\frac{1}{P_a}\nabla P] = 0 \qquad (1)$$

$$\rho=\frac{1}{RT}(P_v\Omega_v+P_a\Omega_a) \qquad (2)$$

$$q=-\frac{K}{\mu}(\nabla P-g\rho\hat{z}) \qquad (3)$$

where q is the volumetric flux of gas (or Darcy velocity), T is the temperature, P is the pressure, P_v is the vapor pressure of water, P_a is defined by $P_a = P-P_v$, ρ is the gas density, R is the gas constant, Ω_a and Ω_v are the molar weights of dry air and water, K is the permeability of the rock, μ is the viscosity of the gas, g is the acceleration of gravity, and $\hat{z}$ is a downward-pointing unit vector. To avoid the numerical problems associated with the subtraction of two large numbers to yield a small number, Amter and Ross[4] recast the problem in terms of a variable h (called "the freshwater head") defined by

$$h=\frac{P-P_o}{g\rho_o}-z \qquad (4)$$

where P_o and ρ_o are reference values of pressure and density. The governing equation then is approximated as:

$$\nabla^2 h - m\nabla T\cdot\nabla h+[\frac{1}{T}+g\frac{\Omega_a-\Omega_v}{RT}\frac{dh_v}{dT}+\rho'(\frac{1}{T}+m)]\frac{\partial T}{\partial z} - \frac{g\Omega_a}{RT}-\frac{\rho'}{h_a} = 0 \qquad (5)$$

where

$$m=\frac{1}{\mu_o}\frac{d\mu}{dT}\Big|_{T_o}+\frac{1}{h_a}\frac{dh_v}{dT}+\frac{1}{T_o} \qquad (6)$$

$$\rho'=\frac{\rho}{\rho_o}-1 \qquad (7)$$

and h_a and h_v are related to P_a and P_v by (4).

A new addition to TGIF is a post-processor particle tracker, which facilitates the direct generation of gas particle path lines. The particle tracker is based on the theory developed by Pollock[7], with modifications for use with a node-centered flow model.

PHYSICAL PROPERTIES OF THE SYSTEM

Gas flow was simulated in a two-dimensional vertical section that cuts across the south portion of Yucca Mountain where a nuclear waste repository might be located. A schematic of the simulated section is shown in Figure 1.

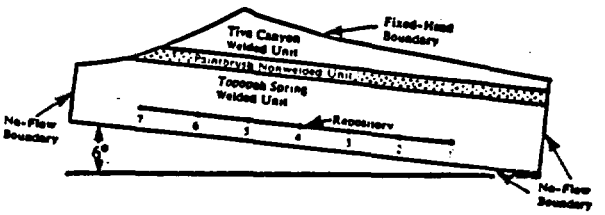

Figure 1. Geometry of cross section used in the gas flow simulation.

The mountain contains a number of hydrostratigraphic subdivisions of the Paintbrush Tuff Formation. These layers dip approximately six degrees to the east and differ in permeability. The most important hydrostratigraphic feature of the modeled section is a thin, nonwelded tuff layer which includes all or part of several stratigraphic subdivisions of the Paintbrush Tuff.[1] This unit, the Paintbrush nonwelded unit, is sparsely fractured and thus is thought to have a relatively lower permeability. It lies between two thick, welded, densely fractured, and relatively permeable units, the Tiva Canyon welded unit (above) and the Topopah Spring welded unit (below).

A permeability of 10^{-11} m^2 is used for both the Tiva Canyon unit and the Topopah Spring unit while permeabilities of 10^{-14}, 10^{-13}, 10^{-12}, 10^{-11}, and 10^{-10} m^2 are used for the nonwelded unit in the simulations. Since the path lines depend only on the degree of the permeability contrast between the two kinds of tuff and not on the absolute magnitude of the permeability, travel times for other values of welded-tuff permeability can be obtained from travel times reported here by dividing by the ratio of the permeabilities.

The simulation region is surrounded by two types of boundary conditions (Figure 1),

- the mountain's atmospheric contact along its surface, and
- no-flow conditions along the base and sides.

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The no-flow boundary assigned at the base of the simulated region represents the top of the low-permeability tuffaceous beds of the Calico Hills unit, which would impede downward gas flow. The boundary to the west is located in the trough of Solitario Wash, which is a natural flow divide. The third no-flow boundary is located far enough to the east to have little effect on flow near the repository, as shown by additional simulations not reported here. Numbers shown in Figure 1 represent starting locations of particle tracks.

The four temperature fields were calculated using Laplace's equation and prescribed temperatures at the repository and all boundaries. Along the atmospheric boundary, a uniform temperature (independent of elevation) was assumed based on an assumed annual average. Temperatures at the base of the cross-section varied with elevation, following an assumed geothermal gradient.

SIMULATIONS

A matrix of 80 runs was constructed by varying three important aspects of the simulation (see Figure 2). These were:

- the temperature of the repository and surrounding rock,
- the permeability of the Paintbrush nonwelded unit, which is the middle layer in the simulation, and
- nodal density used to represent the Paintbrush nonwelded unit and also the entire simulation.

The entire three-dimensional matrix of simulations is shown schematically in Figure 2, where K is the intrinsic permeability of the Paintbrush nonwelded unit and K' is the intrinsic permeability of the surrounding Tiva Canyon and Topopah Spring welded units.

Five different permeability contrasts were used in the simulations. The Paintbrush nonwelded unit was assumed to be 10 times more permeable and also 1, 10, 100, and 1000 times less permeable than the surrounding welded units, which were assigned a permeability of 10^{-11} m^2. The objective was to test the model over the range of contrasts that was found by Montazer et al.[6]

As depicted in Figure 2, the matrix of simulations included four different assumed temperature profiles.[4] The ambient condition represents the current condition of a linear geothermal gradient of 2.0 K per 100 m. In the remaining three situations, the subsurface was assumed to have been heated by the nuclear waste in the repository, raising temperatures 3, 14, and 30 K over ambient conditions. Although these are temperatures that might be expected thousands, tens of thousands, or perhaps hundreds of thousands of years from now, future repository temperatures are uncertain, and these simulations are not intended as predictions of conditions at any particular time. Rather, this range in temperatures was used to examine how gas flow may be affected by temperature.

To examine the sensitivity of the model to mesh density, the number of rows used in the simulation, particularly in and around the middle layer, was varied. Four different meshes were employed.

The first mesh contained 12 rows and 46 columns of blocks. The middle layer contained two rows of rectangular blocks that measured 20 m (vertical length) by 40 m. The remaining blocks were squares with sides of 40 m.

In the second mesh, the number of rows and columns was doubled to 25 and 92. This resulted in a mesh consisting entirely of square blocks that measured 20 m on a side. As in the first case, two rows of blocks were used to represent the Paintbrush nonwelded unit.

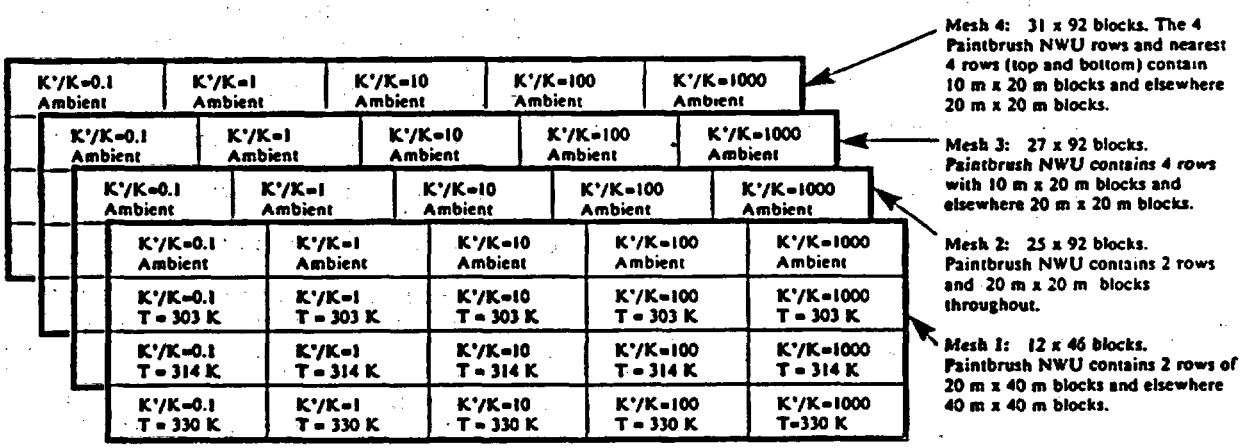

Figure 2. Matrix of simulation cases.

The third mesh was a modification of the second mesh. The number of rows of blocks used to represent the Paintbrush nonwelded unit was increased from two to four by decreasing by half the row spacing in the middle unit. The Paintbrush nonwelded unit was thus represented by 4 rows of rectangular blocks that measured 10 m by 20 m. The rest of the domain retained square blocks.

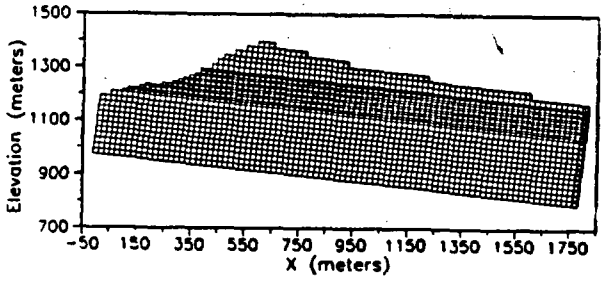

Figure 3. Two-dimensional finite difference grid (31 × 92 blocks).

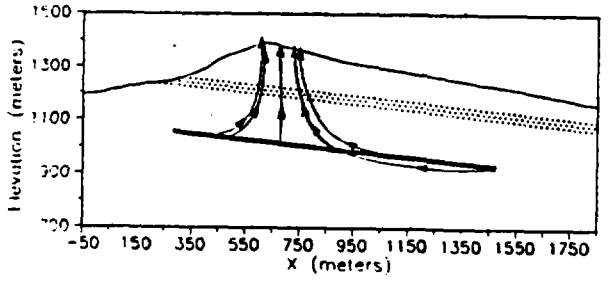

Figure 4a. Path lines with ambient temperature, no permeability contrast.

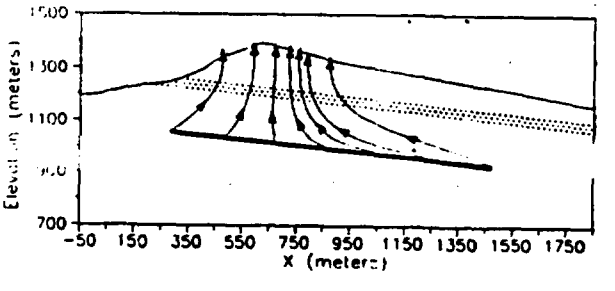

Figure 4b. Path lines with the repository heated to 314 K, no permeability contrast.

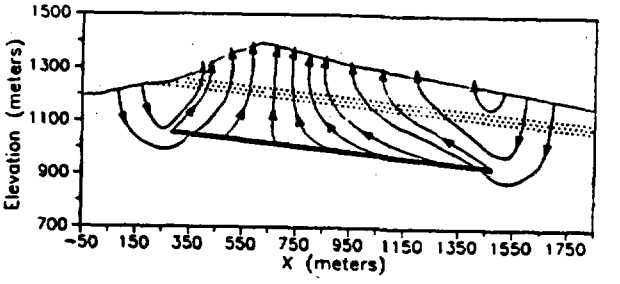

Figure 4c. Path lines with the repository heated to 330 K, no permeability contrast.

The fourth mesh was also a modification of the second mesh. The number of rows of blocks was increased to 32 by halving the row spacing not only in the Paintbrush nonwelded unit, but also four blocks into the surrounding units. This mesh, shown in Figure 3, contained 12 rows of blocks measuring 10 m by 20 m, 4 of which represented the Paintbrush nonwelded unit. The rest of the mesh retained square blocks.

RESULTS

The results of the 80 simulations demonstrate that the predicted pattern of flow is highly dependent on the temperature and permeability contrast. Selected output from the model, in the form of particle tracks, illustrates the major trends.

The Effect of Temperature

The temperature of the repository affects both the velocity of gas particles leaving the repository and the direction of the path lines. Figures 4a-c show the path lines for simulations in a uniform medium (no permeability contrast) at ambient conditions and repository temperatures of 314 K and 330 K.

As the repository temperature increases, the vertical velocity component for gas particles exiting from the repository also increases. Increasing temperature thus decreases path-line curvature and length and also decreases transit times for particles traveling from the repository to the atmosphere. Figure 5 shows how the shortest particle travel time from the repository to the surface is related to the repository temperature. In all simulations, particles starting from near the left end of the repository had the shortest travel time. Since the contrast between the permeability of the middle and surrounding layers also affects travel time, curves corresponding to five different permeability contrasts are shown.

64

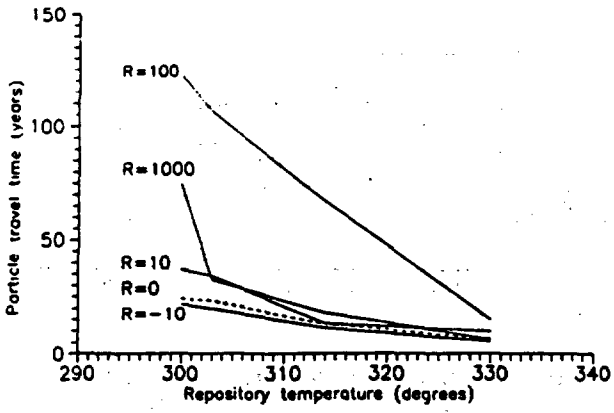

Figure 5. The minimum particle travel time from the repository as a function of temperature at the repository.

The Effect of Permeability Contrast

The degree to which the permeability of the middle unit differs from the surrounding layers has a pronounced effect on the flow system predicted by the TGIF model. When a particle crosses a permeability boundary, its trajectory appears to follow the law of tangents, as one would expect. In addition, temperature and tilting of the layers affect the pattern of gas flow. With sufficient permeability contrast, the middle layer acts as a true confining layer, with independent flow systems above and below it.

Low-temperature situation Figures 6a-e show path lines with the ambient temperature profile (no repository heating) and five different contrasts in permeability. In Figure 6b, the uniform permeability case, the path lines are smooth, are nearly symmetric around the center of the mountain, and all exit near the crest of the mountain. Figure 6a shows the case where the middle layer is 10 times more permeable than the surrounding layers. As would be expected, refraction in the direction of the bedding plane causes greater convergence of the flow lines. Figures 6c-e show that when the middle layer has a progressively smaller permeability, the Paintbrush nonwelded unit becomes an increasingly effective confining layer. In Figure 6c (middle layer 10 times less permeable), path lines are refracted perpendicular to the bedding plane, reducing the convergence of the path lines. In Figure 6d (permeability 100 times less), path lines are diverted some distance laterally outward beneath the middle layer before penetrating it and traveling to the surface. Note that there is a pronounced convergence of these path lines above the layer. At a permeability contrast of a thousand, shown in Figure 6e, confinement by the middle layer is nearly complete; only one gas particle penetrates the layer. Path lines originating from the left side of the repository are directed laterally beneath the middle layer until they exit at the atmospheric boundary. Path lines under the layer from the center and right side of the repository form a convection cell driven by a small horizontal temperature gradient. (When the temperature boundary conditions are changed slightly to eliminate the horizontal gradient, the convection cell disappears.)

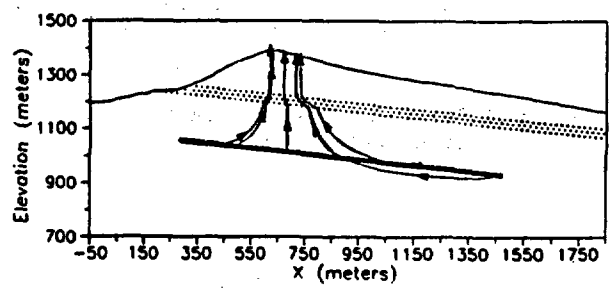

Figure 6a. Path lines with ambient temperature, 0.1× permeability contrast.

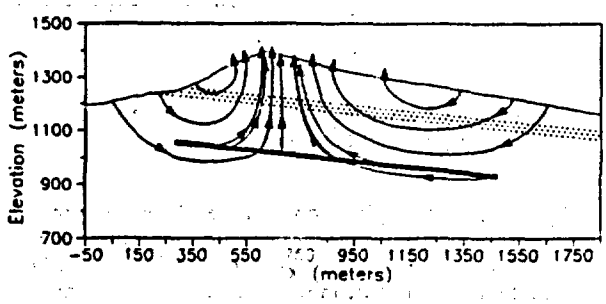

Figure 6b. Path lines with ambient temperature, no permeability contrast.

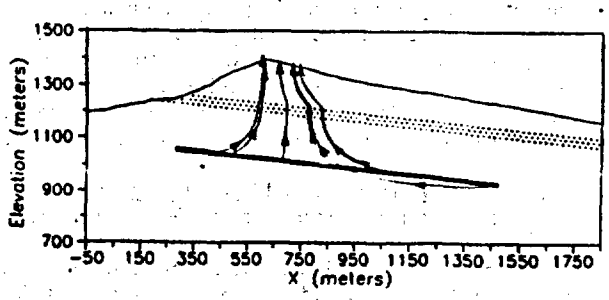

Figure 6c. Path lines with ambient temperature, 10× permeability contrast.

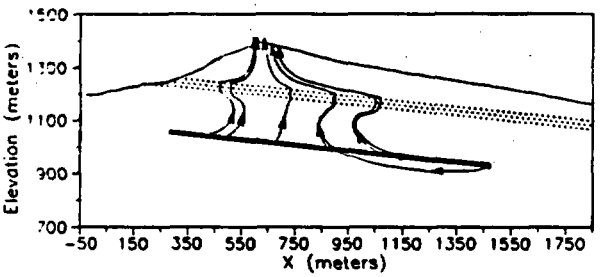

Figure 6d. Path lines with ambient temperature, 100× permeability contrast.

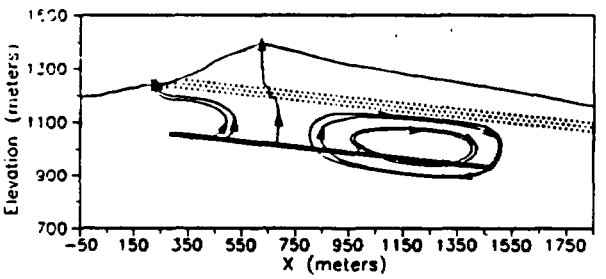

Figure 6e. Path lines with ambient temperature, 1000× permeability contrast.

High-temperature situation Figures 7a-e show the path lines from a repository heated at a temperature of 330 K with five different contrasts in permeability. Although refraction again occurs in the middle layer, there are some important differences from the low temperature situation. Because the higher temperature causes a stronger buoyant flow, path lines tend to be shorter and more vertical when the permeability contrast is relatively small. This can be seen in Figures 7a-b. However, the higher velocities also promote divergent flow beneath the middle layer and the formation of convection cells at both ends of the repository as the permeability contrast increases (Figures 7c-e). When the permeability contrast reaches a factor of 1000 (Figure 7e), the middle layer becomes an effective confining layer and no path lines penetrate it.

Development of an upper and lower flow system As the permeability contrast increases, there is an increased tendency toward the development of separate flow systems on either side of the middle layer. This can be seen by comparing Figures 4c and 7e. In the no-contrast, high temperature case depicted in Figure 4c, the mountain contains a single flow system with a simple pattern of circulation. Some of the air entering the mountain flows deep enough to pass through the repository. All path lines originating from the repository exit near the crest of the mountain.

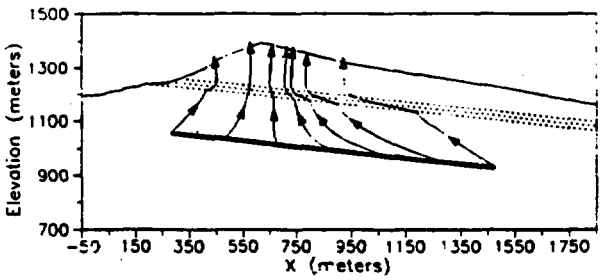

Figure 7a. Path lines with the repository heated to 330 K, 0.1× permeability contrast.

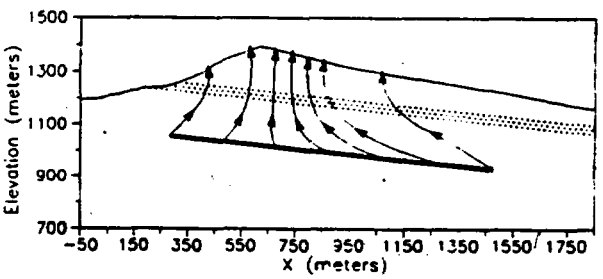

Figure 7b. Path lines with the repository heated to 330 K, no permeability contrast.

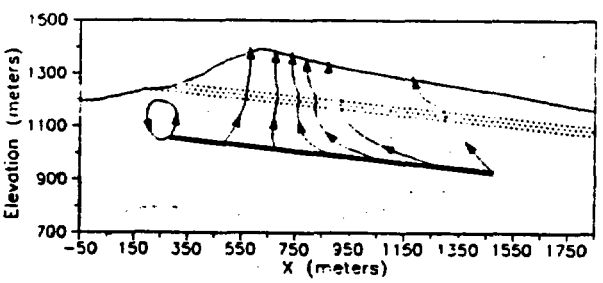

Figure 7c. Path lines with the repository heated to 330 K, 10× permeability contrast.

A very different situation can be seen in Figure 7e, which depicts the high contrast, high temperature case. Completely separate flow systems form above and below the middle layer because gas particles cannot penetrate it. Circulation in the upper system is very shallow and exits at the mountain's crest. In the lower flow system, gas particles from the left portion of the repository exit the left slope of the mountain where

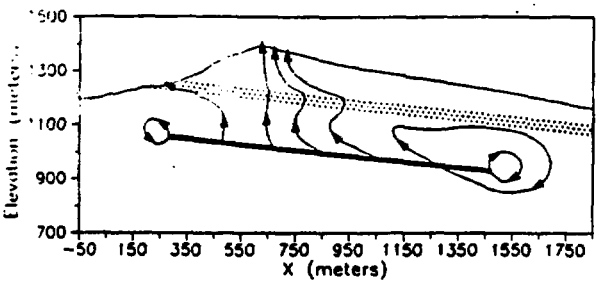

Figure 7d. Path lines with the repository heated to 330 K, 100× permeability contrast.

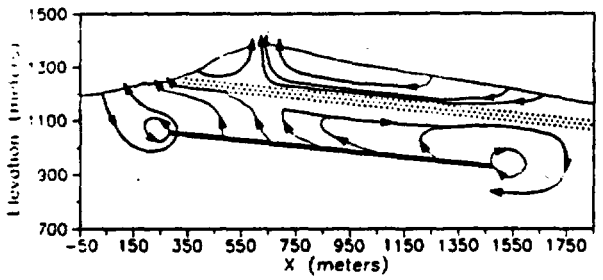

Figure 7e. Path lines with the repository heated to 330 K, 1000× permeability contrast.

the lower layer crops out. Gas particles released from the right portion of the repository circulate back on themselves.

Travel Times Despite the thinness of the Paintbrush nonwelded unit, its permeability has a significant affect on the time required for gas particles originating from the repository to exit the mountain. Travel times generally increase as the permeability of the Paintbrush nonwelded unit decreases. However, reducing the permeability by three orders of magnitude only increases the travel time by one order of magnitude. The reduction in travel times arises from two causes: decreased velocities through the layer and longer path lengths due to refraction. Figure 8 shows the relationship between permeability contrast and the minimum travel time for four different repository temperatures. Up to a permeability contrast of 100 times, there is a steady rise in travel time. An interesting exception to the trend occurs when the permeability contrast is 1000 times. Travel times for particles originating from the left side of the repository decrease because diversion beneath the middle layer causes path lengths to shorten. Examples of this can be seen in Figures 6e and 7e.

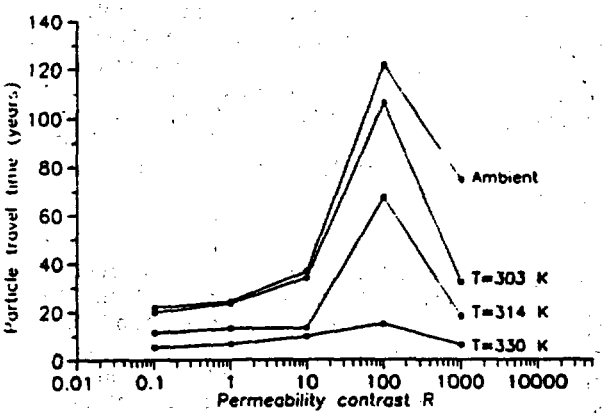

Figure 8. The minimum particle travel time from the repository as a function of the permeability contrast.

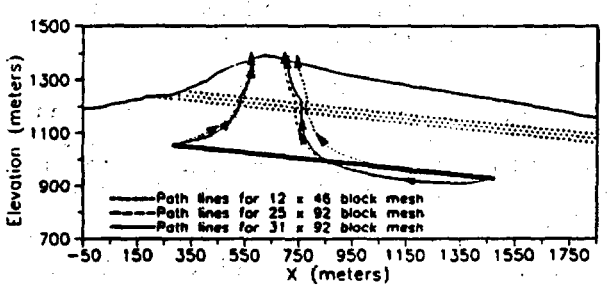

Figure 9. Comparison of path lines for different mesh densities with the repository heated to 303 K, 10× permeability contrast.

The Effect of Mesh Density

Varying the mesh density produces only minor changes in predicted gas path lines. This can be seen in Figure 9, which shows path lines for three different mesh densities when the repository temperature is 303 K and the permeability contrast is 10 times. The predicted path lines for particles released from the repository are very similar for the three different mesh densities. In fact, at the left side of the repository the path lines are virtually indistinguishable. This is also the case for path lines originating from positions 2 through 6. The fact that the path lines are relatively insensitive to the changes in the mesh density indicates that all meshes used in this study are fine enough to yield reasonable results.

CONCLUSIONS

The TGIF model appears to be capable of simulating gas flow at Yucca Mountain over a wide range of inputs. Gas-flow path lines and travel times are highly dependent on the repository temperature as well as the degree of contrast between the Paintbrush nonwelded unit and surrounding layers. At extremely high permeability contrasts, two independent flow systems form above and below the middle layer.

The model can be used to calculate unretarded gas particle travel times in Yucca Mountain. By accounting for retardation, carbon-14 travel time could also be calculated. If the predicted travel time is less than 10,000 years, then more detailed analysis will have to be considered to determine compliance with proposed EPA regulations. Such analysis may require additional, more definitive data describing permeabilities in the mountain. Other parameters, such as waste-package performance, may also need to be considered.

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BAROMETRIC PUMPING OF CONTAMINATED GASES THROUGH FRACTURED PERMEABLE MEDIA

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ABSTRACT

Contaminated gases may be transported vertically through a fractured permeable medium by the breathing process which is associated with cyclical changes in the barometric pressure. A review of results from analytical and numerical modelling indicates that the contaminant transport induced by barometric pumping may be orders of magnitude greater than the rate of transport by molecular diffusion.

INTRODUCTION

Weather patterns cause cyclical variations in the barometric pressure over periods of a few days. As the barometer falls, gases are drawn upward out of the permeable earth into the atmosphere. Conversely, a rising barometer pushes fresh air downward. In a homogeneous permeable medium these vertical gas motions are piston-like and nearly reversible, so they contribute only modestly to the transport of trace gases, as demonstrated years ago by Buckingham.[1] In a fractured permeable medium, however, vertical fractures will serve as the primary breathing passages for all of the underground air volume, causing a substantial increase in vertical gas displacements. The associated breathing mechanism may be orders of magnitude more significant than molecular diffusion in transporting the toxic gases which are released from waste burial facilities.

Experience gained in underground nuclear testing has established a connection between vertical fractures, barometric pumping, and the release of radioactive gasses. Although the associated level of radioactivity has never posed a health hazard, the Nuclear Test Containment Program at Lawrence Livermore National Laboratory (LLNL) is conducting experiments and analysis to better understand the atmospheric pumping process. A partial summary of that work is reported here.

The importance of fractures in carrying the vertical gas flow has been demonstrated by three different suites of atmospheric pumping experiments conducted by LLNL at a total of 20 different sites within the boundaries of the Nevada Test Site (NTS).[2,3,4] Although the details of these experiments have differed, each involved measurements of the pressure history at two locations, one on the surface and the other in a partially cased hole which was capped at the surface The bulk pneumatic diffusivity of the intervening medium was deduced by comparing these pressure measurements with corresponding solutions of the parabolic partial differential equation which governs the isothermal flow of an ideal gas through a homogeneous permeable medium.[5] For alluvial media, the pneumatic diffusivities inferred from these large-scale atmospheric pumping experiments (0.05-2.5 m^2/s) were consistent with core sample and borehole measurements, suggesting that inhomogeneities were relatively unimportant in these media. At volcanics sites, however, there was an enormous disparity between the large-scale bulk diffusivity (0.1-15.0 m^2/s) and the small-scale core-sample diffusivity (~0.00005 m^2/s), suggesting that vertical cooling cracks are the primary breathing passages of these volcanic formations.

The purpose of the present paper is to explain some fundamental features of the barometric pumping process and to present quantitative estimates of the associated contaminant transport, based on numerical and analytical models which are described in two previous publications.[6,7] The results reported here are mostly for moderate weather cycles with a fractional pressure change of 1/100, as opposed to our earlier parameter studies which addressed worst-case scenarios, as often done in evaluating the containment of underground nuclear tests.

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JULY 1975 Scientific American

A Natural Fission Reactor

Two billion years ago in west Africa a rich deposit of uranium ore began operating as a nuclear reactor. Long dormant but preserved intact, the reactor came to light during the mining of the deposit

by George A. Cowan

In 1942, when Enrico Fermi and his associates started up their nuclear-fission reactor at Stagg Field in Chicago, there was every reason to believe it was the first such reactor on the earth. The record book must now be corrected. In an open-pit uranium mine in the southeastern part of the Gabon Republic, near the Equator on the coast of West Africa, are the dormant remains of a natural fission reactor. Within a rich vein of uranium ore the natural reactor once "went critical," consumed a portion of its fuel and then shut down, all in Precambrian times. The experiment at Stagg Field had been anticipated by almost two billion years.

The history of the natural reactor is an extraordinary sequence of seemingly improbable events. First, uranium from an entire watershed accumulated in concentrated local deposits, including one at a place now called Oklo. Then the conditions necessary to sustain the fission chain reaction were established; these included constraints on the concentration of uranium in the ore, on the size and shape of the lode and on the amount of water and other minerals present. After the reactor had shut down, the evidence of its activity was preserved virtually undisturbed through the succeeding ages of geological activity. Finally, the discovery of the reactor involved an investigative tour de force worthy of the best sleuths in detective fiction.

The first clue was found by H. Bouzigues, who is on the staff of the nuclear-fuel-processing plant at Pierrelatte in France. In May, 1972, he obtained a curious result during a routine analysis of a standard sample prepared at Pierrelatte from uranium ore. Natural uranium consists mainly of the isotope of atomic mass 238; only .7202 percent of the atoms are the easily fissionable isotope of mass 235. In the freshly prepared standard Bouzigues found the proportion of U-235 to be even smaller than the usual value: it was .7171 percent.

Bouzigues's analysis was performed by mass spectrometry, in which molecules are ionized and accelerated, then deflected by a magnetic field. The mass of the molecule is revealed by the extent to which it is deflected. The technique is a delicate and precise one, and when it is applied to the gaseous feed materials employed in the enrichment of uranium, it can achieve even greater accuracy than is usual under most other circumstances. The discrepancy Bouzigues had found was a small one, but it was considered significant. The French Commissariat à l'Énergie Atomique (C.E.A.) began an investigation to discover its cause.

The raw material for the isotopic analysis was not the uranium ore itself but uranium hexafluoride gas, the form in which uranium is processed for enrichment. One plausible explanation was that the gas had become contaminated with "tailings," the waste product of the enrichment cycle. The peculiar isotopic composition persisted, however, in the results of repeated analyses of other samples. The possibility of contamination was excluded from further consideration when the anomaly was traced back through the various stages in the manufacture of the uranium hexafluoride gas: through a processing plant in France to the Mounana mill near Franceville in southeastern Gabon. The ore had come from the nearby Oklo mine, operated by the Compagnie des Mines d'Uranium de Franceville (C.O.M.U.F.). Samples had been preserved from each batch of ore processed at the Mounana mill; they showed that shipments of uranium slightly depleted in U-235 had begun in 1970 and were still continuing. By mid-1972 the affected shipments from the Oklo mine involved ore that yielded about 700 tons of uranium; the deficiency of U-235 amounted to roughly 200 kilograms.

The ore body at Oklo had been defined by drilling sample cores on a closely spaced grid. Some of the cores were stored in France, and it was possible to analyze portions of them individually. Several were found to be strongly depleted in U-235. One core, removed from a region of the ore body that was then being mined, contained only .44 percent U-235.

The isotopic composition of uranium is thought to be a constant of the solar system in any one era. (It has been measured not only for many terrestrial ores but also for moon rocks and meteorites.) Chemical processes can make one region rich in uranium and leave another region poor; that is how the deposit at Oklo was formed. U-235 and U-238, however, are virtually indistinguishable chemically, so that any process that affects one of them must affect the other in the same way. Indeed, the difficulty of separating the isotopes is attested to by the size and complexity of uranium-enrichment plants such as those at Pierrelatte and at Oak Ridge, Tenn. There seemed to be no plausible mechanism in nature that might selectively remove one isotope to the extent observed in the depleted ore.

As the investigation continued, the possibility emerged that the missing U-235 had not been displaced but had simply been destroyed in situ. When the required tests were made, the explanation was suddenly obvious. Elements that are characteristic products of nuclear fission were abundant in the depleted vein, but they were almost absent elsewhere in the ore body. Their isotopic composition was quite unlike that of the natural elements, and it corresponded to the composition expected from fission. Three months after the investigation had begun in earnest the mystery was solved. Nature, not man, had constructed the world's first nuclear-fission reactor. Eventually six reactor zones were identified in the Oklo pit, four of them in strata that had not yet been mined.

The fissioning of a U-235 nucleus begins when it absorbs a neutron. The absorption of the neutron excites the nucleus and changes its shape so that about 85 percent of the time it becomes unstable and splits into two fragments and typically two or three neutrons. If at least one of the neutrons is absorbed by another fissionable nucleus and leads to fission, the reaction is

OPEN-PIT URANIUM MINE at Oklo in the Gabon Republic contains scattered pockets of ore that in Precambrian times achieved all the conditions necessary for a fission chain reaction. Six of these "reactor zones" have been identified; part of one, designated Reactor Zone 2, is visible at the left, at the base of the nearer wall of the pit, opposite a stack of core samples draped with a blue tarpaulin. The benches and rostrum, decorated with palm fronds, were set up on the floor of the pit mine for an international meeting convened a year ago to discuss scientific aspects of the reactor.

self-sustaining. The fragments produced by fission are almost always unequal in size—a pair might have masses of 99 and 133—and they are themselves unstable, or in other words radioactive. They decay with half-lives ranging from a few seconds to many years and yield a varied spectrum of daughter fission products. The final products include numerous stable isotopes of more than 30 elements.

The spectrum of fission products is so distinctive that it serves as an unmistakable sign that a chain reaction has taken place. In the Oklo uranium deposit the presence of these elements is convincing and quantitative evidence that a natural reactor once operated there. Both the absolute amounts of the elements and their isotopic composition can be explained only by their origin in fission.

Remarkably, at least half of the 30-odd fission-product elements have remained immobilized in the ore. These include the rare-earth elements lanthanum, cerium, praseodymium, neodymium, europium, samarium and gadolinium and also yttrium, which is not one of the rare earths but is chemically similar to them. Most or all of the zirconium, ruthenium, rhodium, palladium, niobium and silver remains, and so does some of the molybdenum and iodine. There are even remaining traces of the inert gases krypton and xenon.

Metals with a valence of one or two have a relatively high solubility in water and are readily leached away; hence the rubidium, cesium, strontium and barium made in the reactor, and probably the cadmium, have for the most part disappeared. On the other hand, there is no appreciable deficit of zirconium 90; this isotope of zirconium is produced by the decay of strontium 90, which has a half-life of about 30 years. One can conclude that little of the strontium was transported from the vicinity of the reactor

REACTOR ZONE 2 on the floor of the pit at Oklo is marked with pegs and strings, which define the lines along which sample cores were taken. The reactor was discovered in 1972, following the observation in a uranium-processing plant in France of an anomaly in the relative abundance of the two common isotopes of uranium, uranium 235 and uranium 238. The anomaly was traced to ores from the Oklo mine, which were found to be depleted in U-235, the more easily fissionable isotope. Cores from the reactor zones contain the uranium deficient in U-235; in addition they contain elements that are characteristic products of nuclear fission. Elsewhere in the mine uranium has a normal isotopic composition and the fission products are absent. On the rock face above Reactor Zone 2 the vein of uranium ore continues. The black regions are shale containing carbonaceous material; yellow regions are oxidized uranium formed on exposure to the atmosphere.

in that period. Lead, a final product of uranium decay, has migrated to some extent. Other fission products are made in quantities that are small compared with the amounts present naturally, and their fate has not yet been determined.

A particularly suitable element for an analysis of fission products is neodymium. It is not an abundant element and therefore the contribution from its natural background—the amount present from sources other than fission—is small. Moreover, neodymium has seven stable isotopes, ranging in mass from 142 to 150, but only six of them are manufactured as fission products. Fission fragments with a mass of 142 do not decay to neodymium but halt in a stable region of the periodic table at cerium 142. The neodymium 142 present in the depleted Oklo ores is therefore unrelated to fission, and from its amount and the known ratios of neodymium isotopes in natural deposits the background levels of the other six isotopes can be calculated. Subtracting these amounts from the total neodymium in the deposit gives the amount of neodymium made by fission.

For a complete isotopic analysis one more correction is required. Neodymium 143 and neodymium 145 both readily absorb neutrons; they are said to have a large neutron-capture cross section. Neutrons were abundant during the operation of the reactor and as a consequence many atoms of these fission products were altered by neutron capture after their formation. On absorbing a neutron neodymium 143 is converted into neodymium 144; neodymium 145 becomes neodymium 146. This effect perturbs the isotopic ratios, and it is particularly important for the mass-143 and mass-144 pair. Its magnitude can be calculated, however, and its influence can be fully accounted for. When these corrections for natural background level and for neutron capture are made, the abundance ratios of the neodymium isotopes correspond precisely to those measured experimentally in modern reactors of the appropriate type.

Similar analyses can be made of other elements present in the ore. An example is thorium. For every six atoms of U-235 that fission when they absorb a neutron, one simply captures the neutron and becomes an atom of U-236. This nucleus has a half-life of about 24 million years and decays by the emission of an alpha particle (a helium nucleus). When an alpha particle is emitted, the atomic mass of the nucleus is decreased by four and the atomic number is decreased by two. All the U-236 made in the natural reactor has by now decayed into thorium 232, a nearly stable isotope. Thorium is found in the reactor zones and is almost absent elsewhere. Similarly, U-238 in the reactor gives rise to a small quantity of U-237; this decays by a series of alpha-particle emissions to bismuth. At Oklo bismuth is present in quantity only in the reactor zones.

What conditions had to be met to achieve a nuclear reaction in the Oklo deposit? The list of requirements is hardly trivial. The basic requirement is that the flux of neutrons be sustained. On the average, the fissioning of a U-235 nucleus results in the prompt emission of 2.5 neutrons; one of them must be absorbed and must induce fission in another nucleus. The rest can be absorbed elsewhere or escape.

A chain reaction is possible in unenriched uranium containing .72 percent U-235 but only under rather special circumstances. A quantity of deuterium (the isotope of hydrogen with an atomic mass of two) must be present to serve as a "moderator." This is the system employed in the Canadian natural-uranium ("Candu") power reactors. Alternatively, the reactor could be constructed in a geometric lattice of uranium and a moderator with carefully specified dimensions. Neither a deuterium moderator nor a precisely assembled lattice is likely to be found in nature. About the best configuration that has a reasonable probability of being formed accidentally is a mass of relatively pure uranium oxide whose size is large compared with the distance a neutron travels before it is captured. This distance and hence the required size of the deposit are minimized by the presence of an effective moderator. It is reasonable to expect that such a moderator would be present in the form of water saturating the ore.

Given these best plausible conditions, no natural reactor could operate today with uranium containing .72 percent of the mass-235 isotope. The ratio of U-235 to U-238, however, has not been constant throughout the history of the earth. The half-life of U-235 is about 700 million years, that of U-238 about 4.5 billion years. Since the fissionable isotope is decaying faster, it must have been more abundant in the past. Indeed, from the decay rates it is possible to extrapolate into the past to determine the relative abundance of the two isotopes at any time back to the formation of the solar system. When the earth formed, natural uranium was about 17 percent U-235; the isotope has reached its present abundance of .72 percent through an exponential decline. The minimum abundance for the operation of a plausible natural reactor is 1 percent, which means such a reactor could have operated up until about 400 million years ago. In the much older Oklo deposit the relative abundance of U-235 was about 3 percent.

Apart from the isotopic ratio the concentration of uranium in the ore also affects the rate of the reaction; for a chain reaction it must average at least 10 percent. There are also constraints on the shape of the ore body. A sphere is the most efficient shape and requires the smallest quantity of uranium, but it is sufficient that the ore be deposited in seams at least half a meter thick. In a thinner deposit too many neutrons would escape. The reactor zones in the Oklo mine meet the requirements of uranium concentration and seam thickness.

An important requirement for the operation of the reactor is that the neutrons emitted by the fissioning nuclei be slowed down; this is the function of the moderator. The neutrons are emitted with high energy and therefore high velocity; in that state they are readily absorbed by U-238. Ideally neutrons should be slowed to a "therm. distribution of energies: their velocities should be those characteristic of random thermal motion at the temperature of the medium. This can be accomplished if each neutron collides with a great many nonabsorptive nuclei. Slow or thermal neutrons are much less likely to be absorbed by U-238, and hence they are more likely to survive to encounter a nucleus of U-235.

By far the best moderator available in a natural reactor is water. The moderation is provided primarily by the hydrogen atoms in the water, and for an ore that is two billion years old the optimum ratio is about 6 percent water by weight. The water of crystallization in a sedimentary ore such as that at Oklo should more than satisfy this requirement. In addition, at Oklo the medium was probably saturated with groundwater, which would have overmoderated the neutrons. If the ore became chain-reacting in this condition, the heat evolved would evaporate some of the water, so that optimum moderation would eventually be attained. For this reason the constraints on the amount of water initially present are not confining.

Finally, the reactor could not operate in the presence of large quantities of elements that strongly absorb neutrons (that is, elements that have a large neutron-capture cross section). Such elements are called neutron "poisons"; among the more potent ones are lithium, boron and many of the rare earths. There is no evidence that excessive amounts of such poisons were present in the ore before the reaction began.

These circumstances varied during the course of the reaction, changing its rate and eventually stopping it entirely. They are responsible for controlling the power generated, both over short periods and over the entire history of the reactor. Together they functioned to limit the reactor to modest power levels.

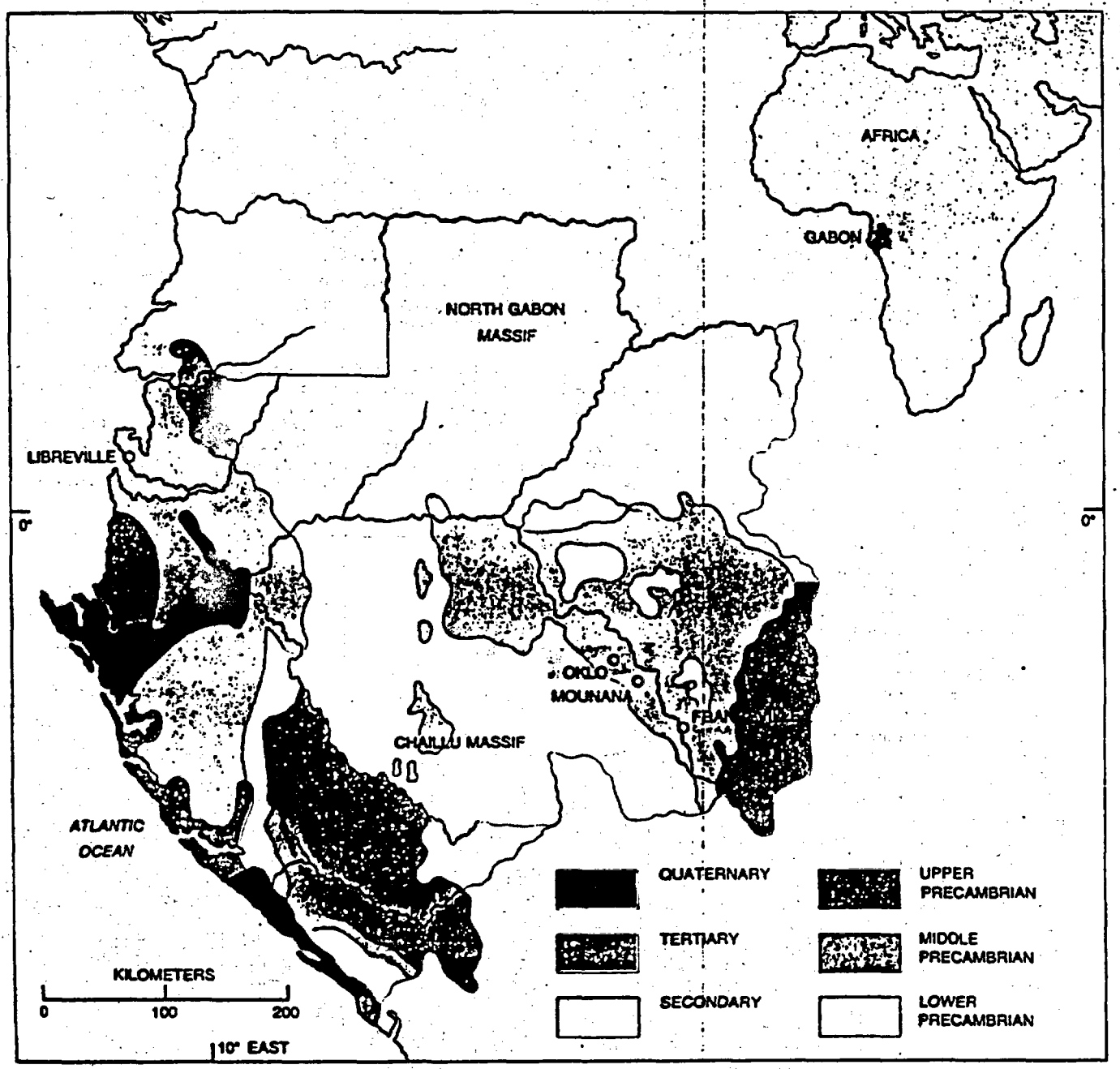
In the long run the behavior of the Oklo reactor was determined by the continuous decrease in the relative abundance of U-235. Almost all this decrease was a direct consequence of the fission itself—the reactor was consuming its fuel—and it was therefore proportional to the total flux of neutrons in the reactor. As I have mentioned, the conversion of neodymium 143 to neodymium 144 (by neutron capture) is also proportional to the total neutron exposure in the ore, and the present isotopic ratios of neodymium therefore predict the total depletion in U-235. This calculation gives a result about 40 percent greater than the observed depletion; in other words knowing the amount of U-235 present initially and the amount remaining, we find that the amount consumed was greater than the difference. Several factors could contribute to this discrepancy, including changes in the concentration or shape of the

deposit during the reaction, and the accretion of ore afterward. The principal cause, however, is an additional nuclear reaction. As we have seen, U-238 readily captures high-energy neutrons; it is thereby converted to U-239. This nucleus decays by the emission of a beta particle (an electron). In beta decay atomic mass remains essentially unchanged but atomic number increases by one. Hence U-239 becomes neptunium 239. Through a second beta emission the neptunium becomes plutonium 239, which decays by alpha emission with a half-life of 24,400 years. The product of the last decay is U-235. Thus the absorption of fast neutrons by U-238 results in the eventual creation of additional U-235.

The operation of the reactor might also have been modified by a decrease in the quantity of neutron poisons present. As soon as a nucleus captures a neutron it is transformed, usually into a nucleus with a smaller neutron-capture cross section. In this way neutron poisons may have been "burned out" of the ore soon after the reactor began operating. If the initial amounts of elements such as lithium and boron were large enough, this effect could have been a major factor controlling the reactor.

Short-term control of the reaction was almost certainly dominated by the amount of water present. If the stratum containing the reactor was buried deep enough, the boiling point of the water may have been 300 degrees Celsius or higher. When the reactor reached that temperature, the water boiled away until the ratio of hydrogen to uranium reached a critical value where the reactor was undermoderated. The power generated by the reactor then leveled off and remained roughly constant, at a level just sufficient to compensate for the heat transferred to regions outside the reactor. (It is possible that the ore deposit was buried so deep that the pressure was great enough to prevent water from boiling, but that has not been demonstrated.)

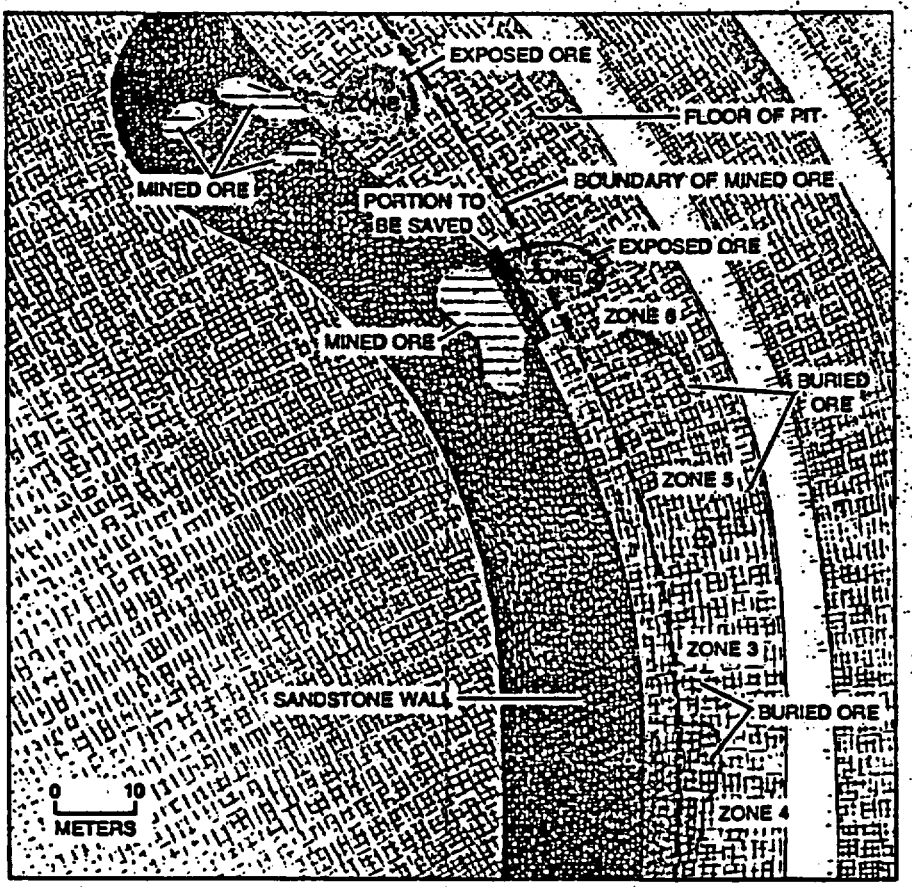
The possibility of a sustained nuclear reaction in a natural uranium ore deposit was first considered more than 20 years ago. In 1953 George W. Wetherill of the University



GEOLOGY of the region surrounding the Oklo mine is predominantly Precambrian, that is, older than about 600 million years. The mine itself is at the edge of a basin made up of sedimentary rock from the Middle Precambrian period; it borders an expanse of igneous rock from the Lower (earlier) Precambrian. The analysis of cores from the reactor zones suggests that the fission chain reaction began between 1.7 and 1.9 billion years ago. Independent evidence gives approximately the same age for the strata in which the reactor is embedded.

of California at Los Angeles and Mark G. Inghram of the University of Chicago stated with reference to a pitchblende deposit: "[Our] calculation shows that 10 percent of the neutrons produced are absorbed to produce fission. Thus the deposit is 25 percent of the way to becoming a pile [a reactor]. It is also interesting to extrapolate back 2,000 million years, when the uranium 235 abundance was [3 percent] instead of .7. Certainly such a deposit would be closer to being an operating pile." Three years later Paul K. Kuroda of the University of Arkansas described the requirements for a natural reactor in a terrestrial uranium deposit in more detail. His description of an "unstable" ore mass comes very close to describing the conditions at Oklo. In spite of such speculations the announcement of the Oklo reactor was received by American nuclear scientists with skepticism. Some of the world's best physicists had constructed the Stagg Field reactor with careful attention to mechanical detail, to the purity of the materials and to the geometry of the assembly. Could nature have achieved the same result so casually?

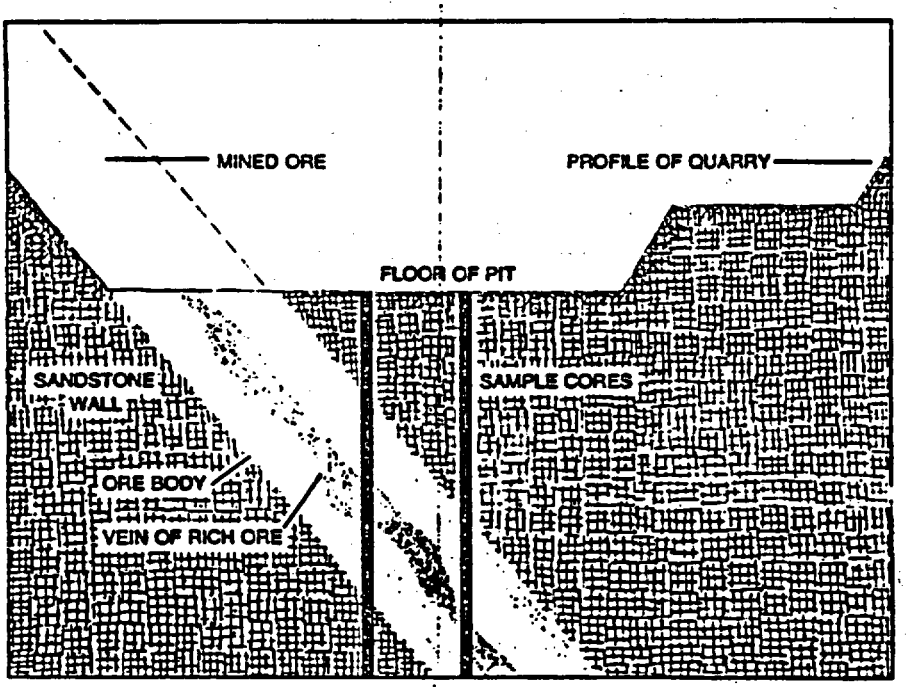
We now know that the answer is yes. Remember again the importance of the difference in date. In the two billion years between Oklo and Stagg Field the relative abundance of U-235 declined from 3 percent to .7 percent. A Precambrian physicist would have found it almost easy to build a nuclear reactor. In retrospect it seems inevitable that it happened accidentally.



SIX REACTOR ZONES at Oklo were found in lenses of exceptionally rich ore. Parts of zones 1 and 2 had already been mined when the reactor was discovered, and their boundaries are therefore conjectural. Zones 3, 4, 5 and 6 have not yet been exposed; they lie below the present floor of the pit and were found by drilling cores. A portion of Zone 2 has been pinned to the rock face so that it will not be destroyed when the pit is deepened. In ore from the reactor zones the average concentration of uranium is from 20 to 30 percent, more than 50 times the concentration elsewhere. On the other hand, the uranium in the reactor zones is strongly depleted in U-235.

Last June 70 investigators from 20 countries met in Libreville, the capital of Gabon, to discuss the "Oklo phenomenon." The meeting was sponsored by the International Atomic Energy Agency (IAEA), the French C.E.A. and the government of Gabon. Its business was to present and review analyses of the Oklo ores, carried out largely in French laboratories but also in the U.S., Britain, the U.S.S.R. and Australia.

The meeting opened with a spectacular expedition: a flight over the tropical rain forest to Franceville in southeastern Gabon and from there some kilometers by car to the floor of the open-pit mine at Oklo. From a rostrum decorated with palm fronds representatives of the sponsoring organizations and ministers of the Gabonese government welcomed us. We sat on benches at the edge of Reactor Zone 2, which was marked out by pegs and strings defining the sampling lines. J. P. Pfiffelmann, chief geologist of the C.O.M.U.F., took up a position at the edge of the reactor zone and lectured on the geology of the formations surrounding us. Behind him a sandstone wall slanted up at an angle of 45 degrees; there were ripple marks on its face attesting to its aquatic origin. The exposed reactor zone, about a meter wide and 10 meters long, extended along the floor of the pit near the sandstone wall. Immediately beyond the reactor zone rose a bench of unmined ore, displaying a continuation of the uranium vein. Water and atmospheric oxygen had converted [illegible] uranium to a [illegible] which formed [illegible]

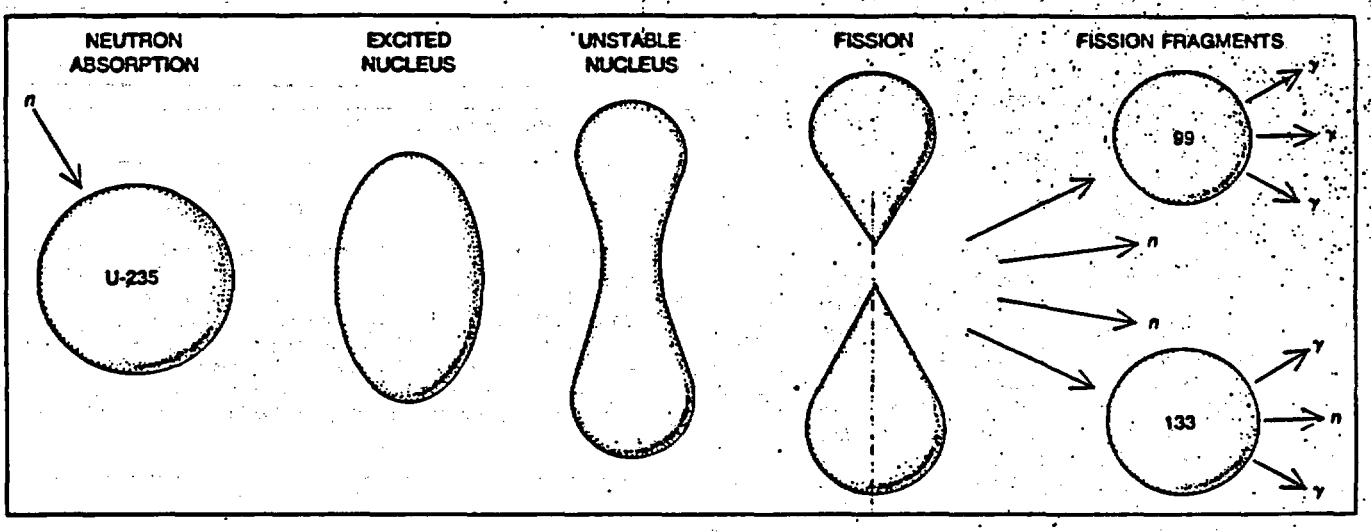


ORE BODY AT OKLO, shown in a schematic profile of the mine, lies above a sandstone wall with an average slope of about 45 degrees. The reactor zones are within the rich vein of ore, which is roughly a meter thick. The zones generally have a lenticular cross section. The extent of the ore body and the location of the reactor zones were revealed by comparing core samples

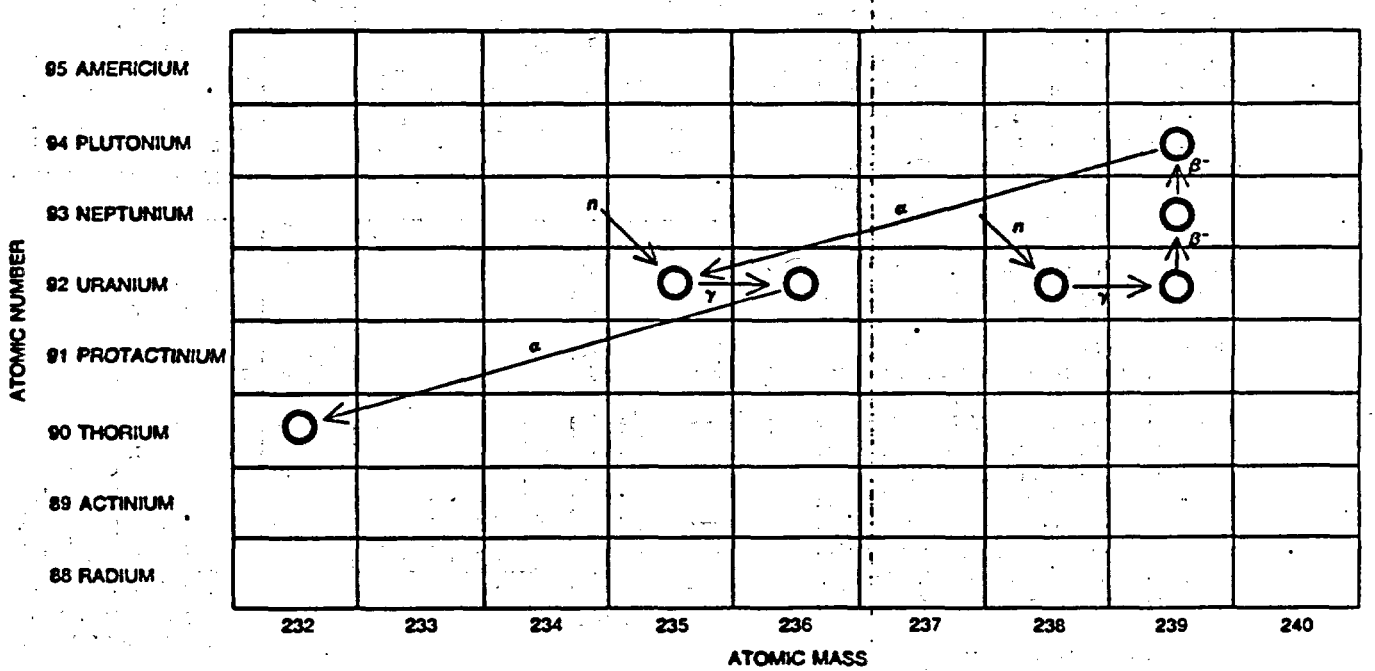
plaques on the buff-colored ore of sandstone and clay.

The landscape around us had once been a river delta. On an ancient African watershed crystalline igneous rocks were eroded by running streams, releasing minute quantities of heavy metals and their oxides. The heavier material accumulated in bottom sediments and pebble conglomerates, much like the placer deposits of gold and other noble metals found in streams today.

About two billion years ago the chemistry of the waters changed abruptly. The change was brought about primarily by the rise of blue-green algae, the first organisms capable of photosynthesis. As a result of their activity the oxygen content of the waters rose, and in some regions it became high enough to convert reduced uranium into more soluble oxidized compounds. The uranium was once again mobilized, and it remained in solution until it reached the river delta. There the bottom sediments were rich in organic ooze and oxygen-poor



FISSION OF A NUCLEUS of U-235 is induced by the absorption of a thermal, or slow, neutron (*n*), which excites the nucleus and deforms it. About 85 percent of the time the deformed nucleus becomes unstable and splits into two fragments of unequal size. The fission fragments shown have atomic mass numbers of 99 and 133; many other pairs of fragments are possible and each has a well-defined probability. The fragments are themselves unstable and are transformed by their subsequent decay, so that the total spectrum of fission products includes many isotopes of more than 30 elements. At the moment of fission high-energy photons, or gamma rays (γ), are emitted, as are a few neutrons. For a chain reaction to be sustained at least one neutron must be absorbed and must induce fission in another U-235 nucleus.



TRANSFORMATIONS OF HEAVY NUCLEI affect the rate and the nature of the fission chain reaction. Most U-235 nuclei that absorb a slow neutron become unstable and fission, but about 15 percent dissipate their energy of excitation by emitting gamma rays and simply retain the neutron. The addition of a neutron does not change the atomic number or the chemical identity of a nucleus, but it increases the atomic mass by one, so that the U-235 becomes U-236. This nucleus is unstable and decays by the emission of an alpha particle (*α*): a helium nucleus, made up of two protons and two neutrons. In alpha emission atomic number decreases by two and atomic mass decreases by four, so that U-236 becomes thorium 232. U-238 readily captures neutrons, particularly if they have a somewhat higher energy than thermal neutrons. It is thereby transformed to U-239, a short-lived species that decays by the emission of a beta particle (*β*⁻): an electron. Beta decay can be considered as the conversion of a neutron into a proton; it does not alter atomic mass, but it increases atomic number by one. The U-239 thus decays quickly into neptunium 239, which in turn decays by a second beta emission into plutonium 239. The plutonium decays, with a half-life of 24,400 years, by alpha emission to yield U-235. Plutonium is readily fissionable, but in the Oklo deposits the rate of the chain reaction was low enough so that most of the plutonium decayed to U-235 before it could absorb a neutron and fission.

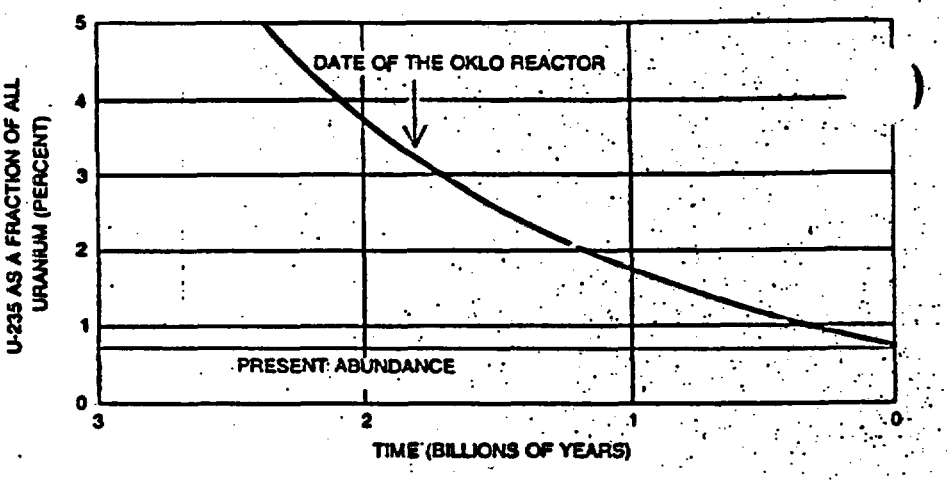
conditions again prevailed. The oxidized uranium was converted into the reduced form and compacted into the sandstone ore that now surrounded us. Subsequently the basement granite sank and the sedimentary layer deepened; then the granite to the west was uplifted, and it tilted the sedimentary overburden to its present angle. The ore layer, which averaged .5 percent uranium by weight, was fractured and water circulated through the new channels, creating pockets of rich ore that in places became almost pure uranium oxide. It was in these rich pockets that fission chain reactions began as soon as a critical mass of uranium had accumulated.

If uranium dissolves so readily in oxygenated water, how has the Oklo deposit survived almost two billion years? The sedimentary basin was apparently buried deep enough to protect the uranium ore from redissolution during most of its history. Only recently (within the past few million years) has the ore horizon approached the surface, where normal prospecting procedures could succeed in identifying it. Thus the series of special circumstances necessary to the discovery of the Oklo phenomenon includes not only those processes that led to the formation of a natural critical mass of uranium but also the unusual geophysical and geochemical conditions that preserved the ore body for almost half the lifetime of the planet and finally brought it to the surface.

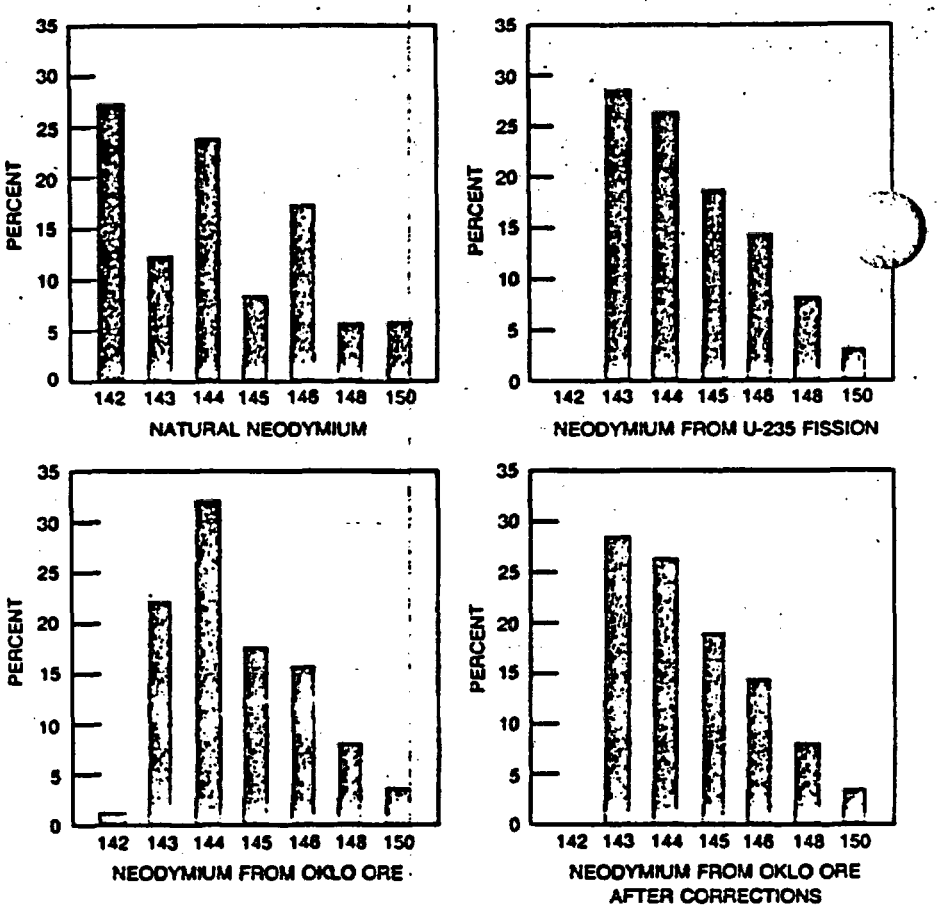
Back in Libreville, Roger Naudet, the director of the C.E.A.'s study of the Oklo phenomenon, reported some of that group's findings. They have estimated that the total energy released in the reactor zones was 15,000 megawatt-years, representing the consumption of six tons of U-235. That is approximately the energy produced by the reactor in a large nuclear power plant in four years.

Only about two-thirds of the fission events involved the U-235 that was originally present. Most of the remainder were in additional U-235 created by neutron capture in U-238 and the subsequent decay through plutonium 239. A few percent of the events were fissions of U-238 induced by fast neutrons, and another few percent were attributed to plutonium 239 that fissioned before it decayed. The modest contribution from plutonium 239 indicates that the reaction lasted much longer than the 24,400-year half-life of that isotope. The duration was on the order of hundreds of thousands of years.

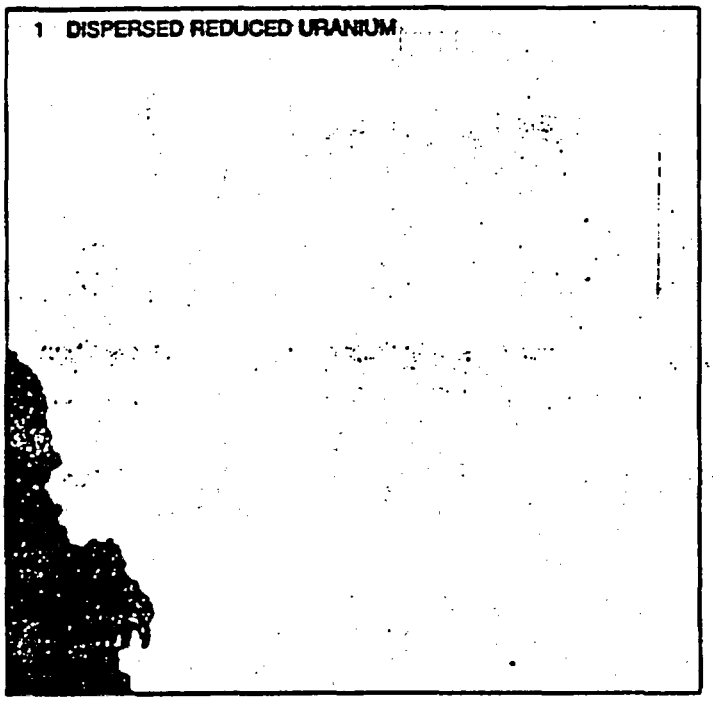
The same conclusion can be deduced from studies of heat transfer in the reactor. Since the reactor could not continue to operate if the temperature much exceeded the boiling point of water, the power level was probably limited to a few tens of kilowatts. This calculation is somewhat uncertain because it is not known how much water was actually flowing through channels in the ore body, but even if the power level was as high as 100 kilowatts, the duration of the reaction would still be 150,000 years. The total

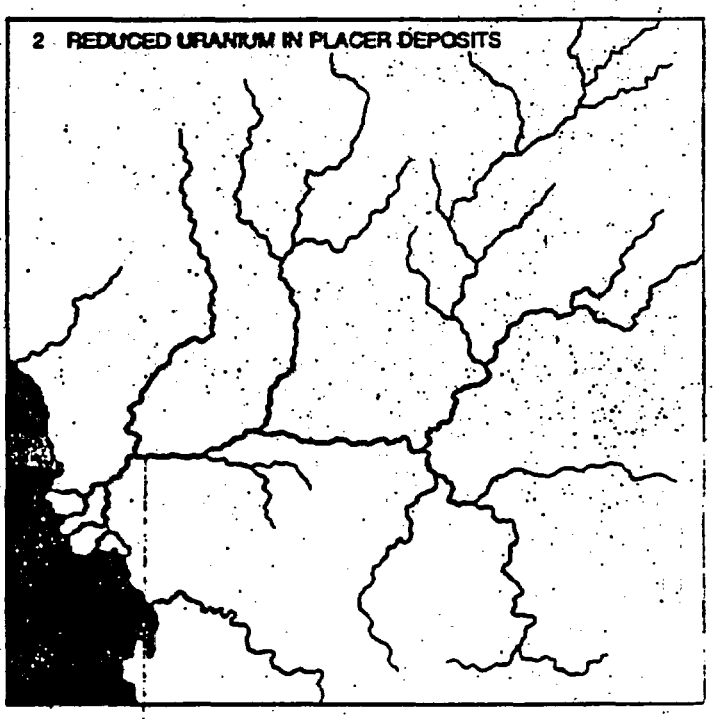


ISOTOPIC COMPOSITION of uranium has changed during the history of the earth. Because U-235 decays about six times faster than U-238 the abundance of U-235 as a percentage of all uranium has declined. It is thought that when the earth formed some 4.6 billion years ago, uranium was about 25 percent U-235; today a commonly accepted value for the relative abundance of U-235 is .7202 percent. When the Oklo reactor was operating, the uranium in the ore contained about 3 percent U-235. The rate of radioactive decay is constant, and at any one moment the isotopic composition of uranium should be essentially the same everywhere in the solar system; it was for this reason the anomalous isotope ratios in ores from Oklo were investigated.



ISOTOPIC ANALYSIS of neodymium, a fission product, provides convincing evidence for a chain reaction at Oklo. Natural neodymium has seven stable isotopes, ranging in mass from 142 to 150, and their relative abundances are a characteristic of the element. Neodymium made by the fission of U-235 has a distinctively different composition; in particular, the isotope of mass 142 is entirely absent. At first the isotopic composition of neodymium from the Oklo reactor zones resembles neither of these distributions, but its origin becomes apparent when two corrections have been made. First, there is a little natural neodymium in the ore, and this must be subtracted. Second, neodymium 143 readily captures neutrons and is thereby converted into neodymium 144; similarly, but to a lesser extent, neodymium 145 is converted into neodymium 146. Since neutrons were abundant in the reactor, the ratios of these isotopes were substantially altered. When the presence of the natural element and neutron capture are taken into account, the composition of the neodymium in the Oklo ores closely matches that produced by fission.

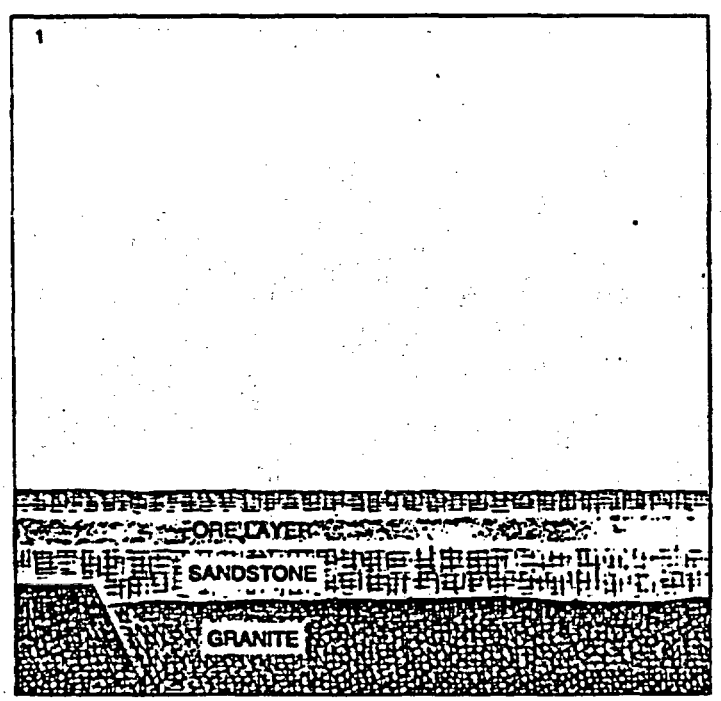


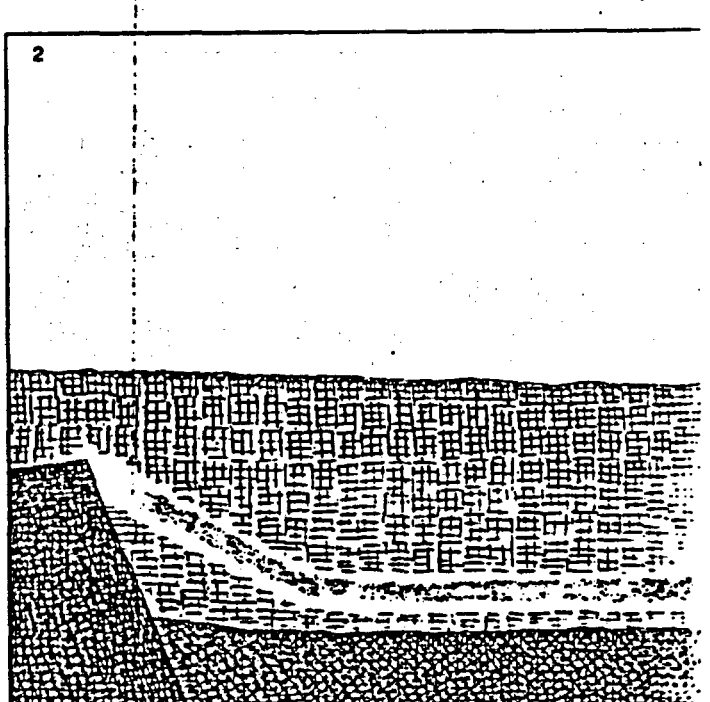


ACCRETION OF THE ORE DEPOSIT at Oklo involved the concentration of uranium derived from an entire watershed. The uranium (*color*) was originally dispersed in igneous rocks over a large area (*1*). As the rocks were eroded by weathering, the uranium accumulated in streambeds in a relatively reduced state (*2*). It may have formed placer deposits like those in which gold and low-grade, Precambrian uranium ores are found today. Some two billion years ago a biological development profoundly altered the disposition of the uranium:

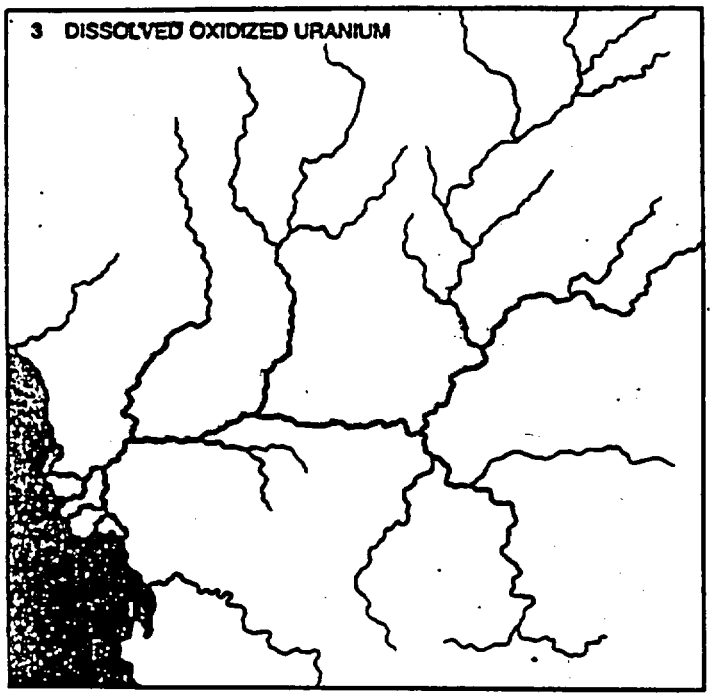
time elapsed from the initiation of the chain reaction to the final shutdown was probably even greater, because the reactor did not necessarily operate continuously.

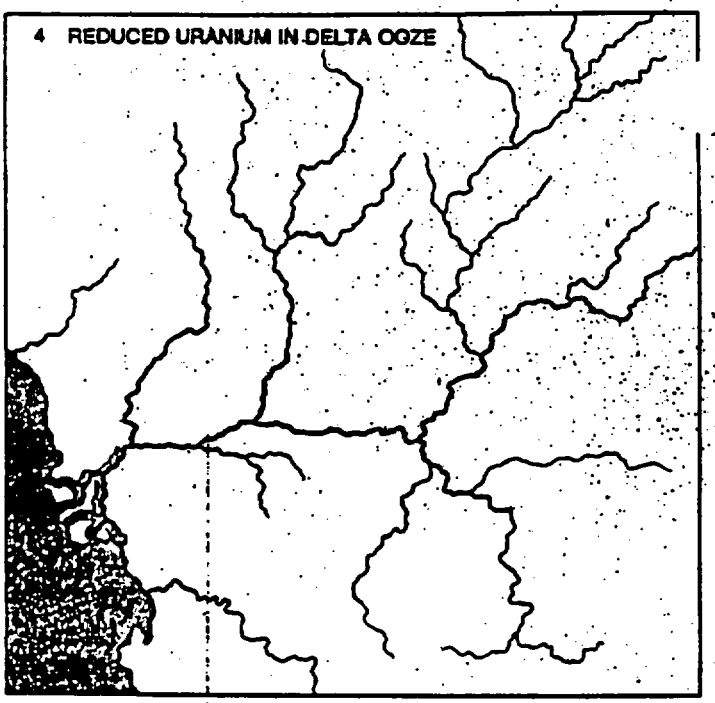
The unusual stability of the ore deposit was confirmed by analysis of samples taken every 2.5 centimeters along several cross sections of the reactor zones. On this scale there was generally good correspondence between the extent of U-235 depletion and the total neutron flux over the life of the reactor, as calculated from isotopic ratios in neodymium. Although some puzzling anomalies were found at the borders, there was little migration of uranium inside the zones. Distinct excursions in both the neutron flux and the U-235 depletion were observed in the vicinity of faults in the ore body. When the reactor was functioning, these faults were most likely water-filled channels that trapped neutrons and increased their local density. Such neutron traps are a feature of some modern research reactors.





GEOLOGICAL TRANSFORMATION of the ore body created the conditions necessary for the operation of the natural reactor and preserved the evidence of the reaction for almost half the life of the planet. After the ore had been deposited in the river delta it was compacted along with other sediments into a layer of sandstone (*1*) overlying basement rock of granite. Gradually the sedimentary overburden deepened and the ore layer sank (*2*), while the granite to the west of the deposit began to be uplifted. As the uplift continued, the vein of ore

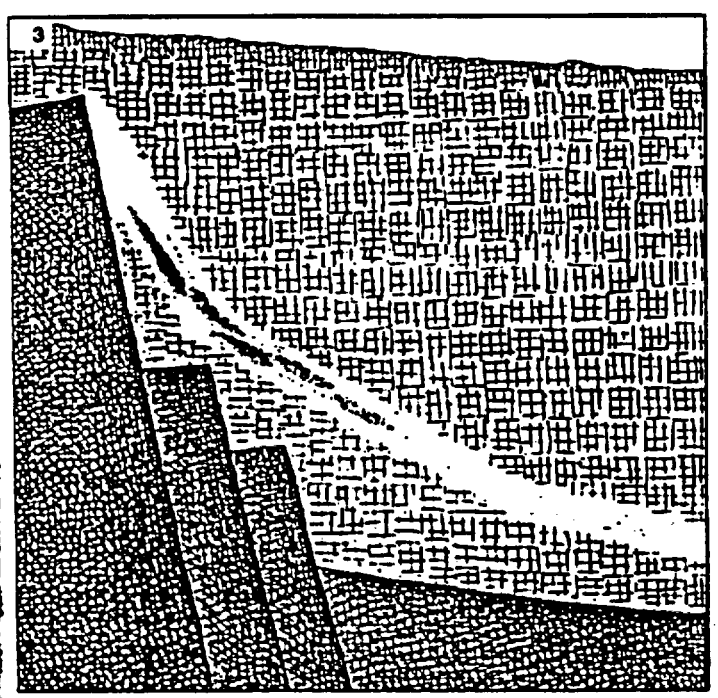


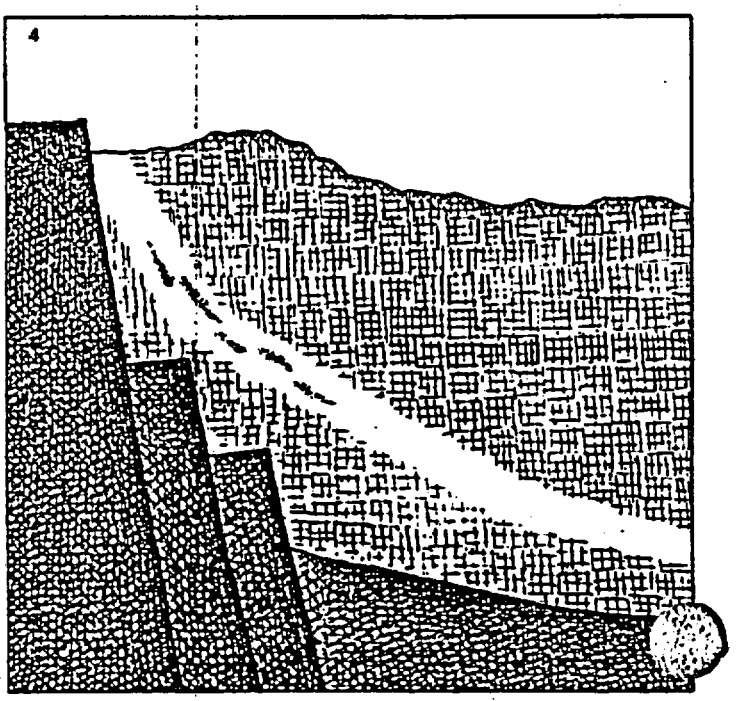


the proliferation of photosynthetic algae increased the concentration of oxygen in the water. Highly oxidized uranium is more soluble than the reduced form, and the uranium was therefore dissolved and carried downstream (*3*). It remained in solution until it reached the delta of the river system, where sediments rich in organic ooze again created a condition of oxygen deficiency. In these sediments the uranium was returned to the reduced form and was precipitated out of the solution (*4*). In the delta the ore was covered by later sediments.

A group of investigators from the University of Paris employed an ion probe to study the distribution of isotopes within individual grains of ore. The partitioning of several fission products between grains of uraninite and clay was noted, but the rare earths and uranium were found in the same grains. More remarkable was the complete confinement of U-235 and U-238 to the same grains. Almost half of the residual U-235 in the reactor zones is a daughter product of plutonium 239. The fact that no regions were enriched in this daughter product indicates that the precursor plutonium was completely immobilized for times comparable to its half-life of 24,400 years. The possible relevance of this observation to proposals for the long-term storage of nuclear wastes was pointed out by investigators from both France and the U.S.

Finally, both French and American participants presented estimates of the age of the reactor based on analyses of the ore samples. The estimates assume that the





was tilted to its present average slope of about 45 degrees. The tilting caused numerous fractures, so that water could percolate through the ore, creating small pockets of very concentrated ore (*3*). In these pockets the chain reaction began as soon as enough uranium had accumulated. For most of its history the ore was probably buried deep under other sedimentary rocks where it was protected from dissolution and dispersal by groundwater. Only much later, in the past few million years, did it approach the surface where it could be detected (*4*).

abundance of uranium and neodymium in the ores has not been extensively altered since the reactor stopped functioning. The absolute abundances of those elements and their isotopic compositions then give a date for the initiation of the reaction. The best fit to the American data is an age of from 1.7 to 1.9 billion years, which is in good agreement with independent estimates of the age of the host geological formation.

Outside the conference, discussion was given over to freewheeling speculation. Did prokaryotes (living cells without nuclei) evolve into eukaryotes (cells with nuclei) as early as 1.8 billion years ago? Was it possible that Africa was not only the cradle of man but also the birthplace of the cells that led to all the higher forms of life? There was general agreement that the plot should be worked into a science-fiction story but that without considerable embellishment the motion-picture rights would be worthless. It would have been far better to introduce a spaceship from another planet that had dumped its used reactors at the site, replenished its fuel supply and departed.

For the three years from the discovery of the depleted uranium to the symposium in Libreville, the C.O.M.U.F. had suspended mining of the very rich ores in the reactor zones. Now it was time to deepen the pit and mine the uranium in the protected regions. Presumably the highly depleted ores would be processed separately. It was proposed to save a portion of Reactor Zone 2 by walling it off and pinning it to the 45-degree sandstone face. The ore will be suspended above the new floor of the pit, an elevated monument to the first discovery of a natural fission reactor.

Clearly it would be interesting to know if there have been other natural reactors of the Oklo type. The final stage of the process that formed the Oklo lode could have been a very common one in Precambrian times: as the oxygen content of the atmosphere rose, reduced uranium in scattered deposits would have become mobile, and it could then have been reconcentrated in richer ore deposits wherever a reducing environment was encountered. Rich uranium ore deposits have been found in other geological formations of approximately the same age, not only in Africa but also in other parts of the world, particularly in Canada and northern Australia. None of these deposits has yet been identified as a reactor site. It is entirely possible that chain-reacting ore lodes formed in these areas and have since disappeared. They may have been buried under younger sediments, where they are unlikely to be discovered, or they may have been dispersed as a result of geophysical instabilities or geochemical mobility.

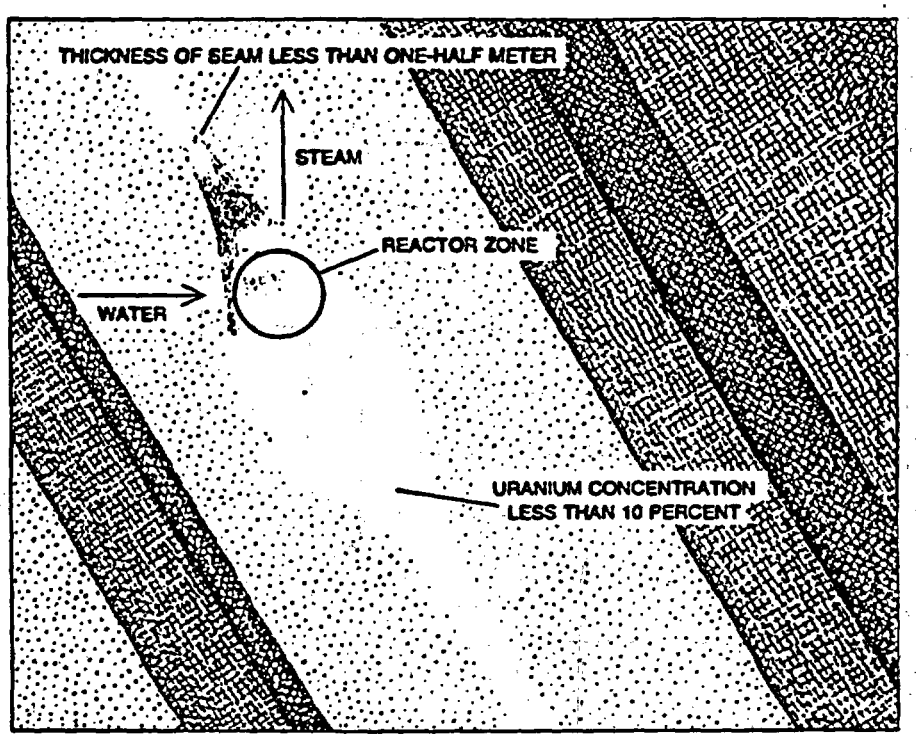
What is the probability of such dispersion? Apart from its obvious scientific interest, the answer to this question has possible economic and technological consequences. It has a direct bearing on the problem of the long-term storage of nuclear wastes. In this regard the stability of the Oklo deposit is plainly encouraging. It contains all the chemical elements of interest and demonstrates that, at least in this one environment, they remained in place for an enormously long time.

At the Libreville symposium Ray D. Walton, Jr., of the U.S. Energy Research and Development Administration (ERDA) presented the results of a preliminary American attempt to determine the significance of the Oklo phenomenon for radioactive-waste storage. He suggested that at the time of the reaction the principal radioactive products that were released into the environment in measurable amounts were krypton 85 and possibly cesium 137 and some strontium 90. Plutonium, the most worrisome reactor product, was efficiently confined.

The adsorptive properties of shales and clays and the extreme stability of many elements bound in such matrixes were known before the discovery of Oklo, and suggestions for the storage of reactor products in such formations have been under investigation for many years. It is obviously difficult to devise an experiment that would evaluate the stability of an adsorbed metal ion for a million years or more under field conditions. Since Oklo resembles such an experiment, the data have been examined with great care to determine what part of them might be relevant to the problem of waste storage.

The possibility that other ore deposits once supported chain reactions also affects the producers and buyers of uranium. It should be of interest as well to those regulatory agencies whose responsibility it is to account for all fissionable materials; they are expected to notice mysterious shortages of a few hundred kilograms of U-235.

In this regard it is interesting to note that not all natural reactors would necessarily give rise to depleted ores. As the isotopic abundance of U-235 diminished over geologic time, the conditions necessary for reactor operation became more restrictive, but at the same time the nature of the chain reaction was subtly altered. In particular, the relative importance of neutron capture in U-238 increased, since that isotope came to form a progressively larger fraction of the total uranium. If a natural reactor was able to form as late as 800 million years ago, when the relative abundance of U-235 was about 1 percent, it might actually have become a breeder reactor. The U-235 consumed in the reaction would have been more than replaced by new U-235 created by the decay of plutonium. The pitchblende deposits in the former Belgian Congo were



RATE OF A CHAIN REACTION is determined by the flux of neutrons, and at Oklo this was influenced by several factors acting in combination. The reaction could begin only in regions of the ore body where the concentration of uranium was greater than about 10 percent. The shape of the deposit was also important: thin seams allow too many neutrons to escape. Elements that act as neutron "poisons," strongly absorbing neutrons, can also prevent a reactor from functioning, but apparently none of these elements was abundant at Oklo when the chain reaction began. A final requirement is a moderator, a substance that slows neutrons so that they are not too readily captured by U-238. At Oklo the moderator was water, and the amount present was presumably the most important factor, over relatively brief periods, controlling the power level of the reactor. Any increase in power would raise the temperature and boil off water, slowing the reaction. Through this mechanism power output was maintained at a modest level. In the long [illegible] by the present-day abundance of U-235

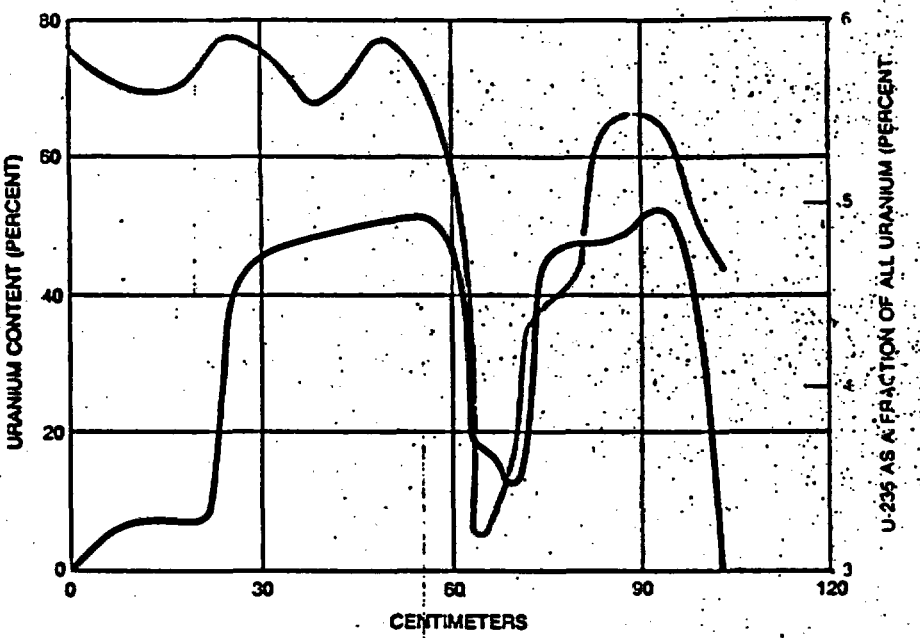
of the appropriate age. They have now been mined out, but precise isotopic analyses are available for a few samples of the ore. It is provocative that these samples appear to be slightly enriched in U-235.

The number of economically significant deposits of any mineral is determined by the difference between the rate at which the deposits form and the rate at which they are dispersed. The second term in this equation—the one dealing with rates of dispersion—has largely been overlooked. Its importance to waste storage is apparent. Ultimately it may also determine the probability of discovering other natural reactors.

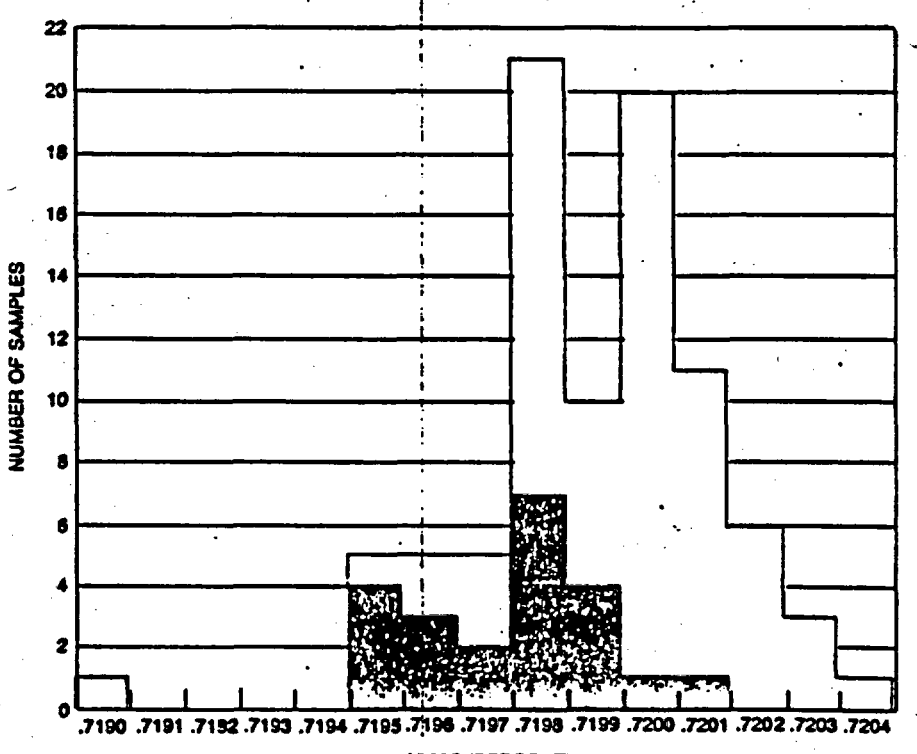
The statement that the isotopic composition of uranium is constant throughout the solar system is based mainly on analyses accurate to about .1 percent. At that level of precision the ratio is indeed constant. The more refined mass spectrometry possible at uranium-enrichment plants gives results accurate to about .01 percent, and at this level variations in the ratio have been discovered. In particular, sedimentary deposits from the Colorado Plateau in the U.S. are depleted in U-235 by about .03 percent. These variations have generally been attributed to chemical differentiation of the isotopes in sandstone rocks. Another hypothesis is possible: the depletion might result from the operation of a reactor of the Oklo type in the Colorado Plateau, the remains of which have since been dispersed throughout the region. The total uranium reserves of the plateau are on the order of 400,000 tons; to explain the observed depletion requires a reactor that caused a net loss of five tons of U-235. We do not yet have enough information to decide which of these explanations is the more likely to be correct.

If dispersion was not the general fate of natural reactors, we can hope to find their dormant remains by surveying rich uranium deposits more than a billion years old. In principle it would even be possible to recognize a reactor zone in which most of the uranium had been dissolved and washed away, since the less mobile rare earths would probably remain in place. Any ore pocket containing rare earths, yttrium, zirconium, niobium, ruthenium and rhodium, all in abundances greater than .01 percent, could be considered a candidate. The origin of these elements in a fission reaction could be demonstrated by isotopic analysis.

On balance, the prospects for finding additional reactors seem good. If Oklo had been a unique event in the history of the earth, the probability of our having discovered it must approach zero. Considering the almost accidental way its existence was revealed, one is tempted to conclude that similar reactors have already been mined out without being noticed. In time we shall learn whether to regard the survival of the Oklo deposit as a unique phenomenon in natural history or as a particularly valuable experiment in long-term geological storage. In any case one message is already clear. In the design of fission reactors man was not an innovator but an unwitting imitator of nature.



STABILITY OF THE OKLO DEPOSIT during the 1.8 billion years since the reactor operated is suggested by a detailed examination of the ores. In some regions samples were analyzed every 2.5 centimeters. (The horizontal axis gives distance along the sampling line.) Generally the concentration of uranium in the ore was closely correlated with the depletion in U-235, as at the extreme left. The sharp dip in uranium concentration is caused by a crack in the ore sample. When the reactor was operating, the crack was apparently filled with water and served as a "neutron trap" that slowed neutrons and greatly increased the efficiency of the reaction in its vicinity. That increased efficiency is reflected in the corresponding decline in the abundance of U-235. The correlation of the two curves suggests there has been little migration of uranium.



ABUNDANCE OF U-235 in 88 samples of uranium ore shows a clearly bimodal distribution. The peak at .7200 percent represents ores mined mainly in Canada, Europe and Australia. The peak at .7196 percent is produced by ores taken from the Colorado Plateau region of the U.S. One group of these ores, mined in Northeastern Arizona, is shown separately (*dark color*); it plainly suggests that uranium from the Colorado Plateau has a U-235 content smaller than the world average. One possible explanation of this small discrepancy is that the isotopes were separated by some chemical process in the ore-bearing rocks. Another explanation is that a natural reactor once operated in the region and that its depleted ore has been dispersed through part of the plateau.

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Natural Analogues for the Transuranic Actinide Elements: An Investigation in Minas Gerais, Brazil

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ABSTRACT / A highly weathered deposit of thorium and rare earth elements located near the summit of a hill in the state of Minas Gerais, Brazil, is being studied as an analogue for a radioactive waste repository that sometime in the distant future may be eroded to the surface or intruded by groundwater. Thorium serves as an analogue for Pu^{4+}, and La^{3+} as an analogue for Cm^{3+} and Am^{3+}. The mobilization rate constants of the analogue elements by groundwater are so slow (of the order of 10^{-9} per year) as to suggest that essentially complete radioactive decay of the transuranic actinides would occur in place even under the relatively unfavorable conditions that exist at a site such as this.

Introduction

An urgent question associated with management of the nuclear fuel cycle is how ancient residues of actinide elements would behave if a nuclear waste repository should be exposed by erosion or be intruded by groundwater. A substantial literature has accumulated concerning the behavior of the transuranic actinide elements in the environment (Watters and others 1980), but most of the research until now has been concerned with fallout from nuclear weapons tests or with contamination in the environs of major atomic energy production plants (IAEA 1976; Hanson 1980; Wrenn 1981). Such studies, useful as they are, leave some questions unanswered because the contaminants have chemical and physical forms different from those that would exist in a nuclear waste repository. Moreover, the fact that these artificial elements were first produced only 40 years ago makes it impossible to consider the effects of geochemical processes over the necessary span of time.

The system of radioactive waste management that is currently most favored involves converting the waste into an insoluble form such as borosilicate glass, packaging the solidified waste in corrosion-resistant canisters, and placing the waste package in deep underground mined cavities, where the wastes must remain isolated from the biosphere long enough to allow decay of most of the radionuclides (NRC 1983). However, many of the nuclides have such long half-lives (i.e., about 25,000 years for ^{239}Pu) that the repository might be breached by groundwater or, in extreme cases, might erode to the surface, before the radionuclides have decayed sufficiently. The potential risks to the public health in the event of such failures would be determined to a considerable extent by the rates of mobilization, movement in groundwater, and uptake by biota of the transuranic actinide elements, plutonium, americium, curium, and neptunium.

To test whether construction of a proposed repository would entail undue risk to public safety, transport models have been developed that predict the dose to humans in the event of repository failure under a variety of hypothetical circumstances (Denham and others 1973). The dose estimates predicted by those models are subject to uncertainties that may span several orders of magnitude.

Naturally occurring elements that have properties similar to those of the transuranic actinide elements offer a possible means of constructing transport models that are more reliable than those now in use. Two conditions must be satisfied to make such models useful. First, the chemical elements in nature must be shown to be valid analogues for the elements of concern. Second, enough must be known about the chemical and physical forms in which the analogue elements occur, and the geochemical processes involved in their mobilization and transport by environmental processes, to ensure that the information obtained is applicable to proposed repository sites.

An unusually favorable site for such studies exists at the Morro do Ferro (MF) in the state of Minas Gerais, Brazil,

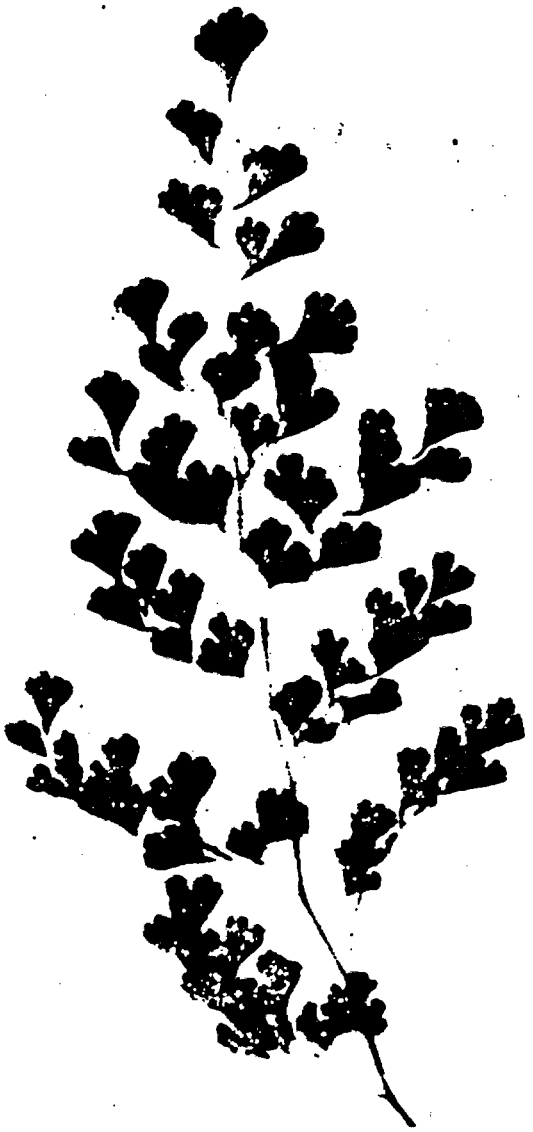

Figure 1. Autoradiograph of a fern, *Adiantum lorentzii*, Hiern. growing on the Morro do Ferro. The radioactivity is due to the uptake of ^{228}Ra, a nuclide in the ^{232}Th decay chain.

where an ore body is located that is rich in Th and rare earth elements (REE) that can be studied as analogues for three transuranic actinide elements. Th is analogous in its behavior to Pu^{4+}, the oxidation state in which Pu exists under many environmental conditions. Nd^{3+} or La^{3+} can serve as an analogue for both Am^{3+} and Cm^{3+}.

General Features of the Morro do Ferro

The MF is a hill near the center of the Poços de Caldas plateau about 300 kilometers north of the city of São Paulo. The ore body, chiefly confined to the south face of the hill, was discovered in the early 1950s. Estimates of the Th, U, and REE content of the ore body were made independently by Wedow (1967) and Frayha (1962), based on analyses of cores. Some of these cores, the spoils from nine wells recently drilled, and a new core that has been drilled through the ore body to a depth of 450 m, are currently being analyzed, using the more advanced methods now available. As an interim working estimate of the Th content, we use a value of 30,000 tonnes, based on the estimate made by Frayha, whose studies included samples from greater depths than were available to Wedow.

The ambient gamma radiation levels near the summit of the hill range from 1 to 3 mR/hr (100 to 300 times normal) over an area of about 30,000 m^2 (Cullen 1977). The plants on the MF are so radioactive (because of absorption of ^{228}Ra from soil) that they can be autoradiographed (Fig. 1), and the exposure of indigenous burrowing rodents to thoron decay products is estimated to result in an average dose to the basal cells of rat bronchial epithelium of about 3,000 rem per year (Drew and Eisenbud 1966).

The Poços de Caldas plateau, which is roughly circular and about 35 km in diameter, is believed to be a deeply eroded caldera. The MF is near the center of the plateau, rising 140 m above its immediate surroundings. The geology of the area has been described by Wedow (1967), Frayha (1962), Almeida (1977), Ellert (1959), and Bushee (1971). Age determinations by Bushee show that intrusion of molten rock took place over a long period: phonolites and tinguaites give ages between 75 and 87 million years, and foyaites 63 to 64 million years. Thus, the body of alkalic igneous rocks was built up by piecemeal additions over some 20 million years toward the end of the Cretaceous Period. The caldera must have formed later by collapse of the intruded rocks.

The underlying rock of MF shows extensive alteration, due in part to weathering and in part to hydrothermal activity. Material fresh enough for identification is tinguaite (fine-grained nepheline syenite), partly massive and partly brecciated. The only rock outcrops at the MF are magnetite, which occurs as a set of subparallel dikes up to a few meters in thickness on the south face of the hill.

The Th and REE are widespread in the surface material and are especially concentrated near some of the dike contacts. However, the Th-rich material is apparently not directly related to the magnetite. From the studies of Frayha and our recent examination of the gamma ray log of the 450-m drill hole, we conclude that abnormal concentrations of Th exist to depths of nearly 200 m. The reason for the concentration of Th and REE at this one site is a mystery. These elements commonly show slight enrichment in alkalic rocks, but the large concentrations at MF are extraordinary.

Wedow (1967) reported that bastnaesite, thorogummite,

able 1. Logarithms of formation constants for thorium and plutonium complexes. Ionic strength 0 except when indicated otherwise in parentheses. (See text for explanation and references.)

Complexing Agent:	F^-	$SO_4^{2-}(k_1)$	$SO_4^{2-}(k_2)$	$HPO_4^{2-}(k_1)$	$HPO_4^{2-}(k_2)$	$C_6H_5O^{3-}$	EDTA
Pu^{4+}	8 ± 1	6 ± 1	10 ± 3	13 ± 1	24 ± 1	15.0 (0.5)	25.7 (0.5)
Th^{4+}	8.0	5.4	9.7	10.8 (0.3)	22.8 (0.3)	13.0 (0.5)	25.3 (0.1)

and cerianite were present, but only as secondary minerals associated with magnetite outcrops. In our previously reported study (Eisenbud and others 1982) of thin sections of some of the deeper cores, zircon, monazite, and cheralite, a rare monazite-like mineral, were found to be present in micron-size crystals. However, this information came from cores that were drilled by wet methods about 30 years ago. It now appears that only a minor fraction of these elements is contained in these highly insoluble minerals, and that most of the thorium is associated with claylike material that was washed away in the process of wet drilling. Samples of rich ore had, therefore, not previously been available for mineralogical study. Nine wells were recently drilled dry, and a study of the spoils, which contain ample rich ore, has shown that most of the Th (and presumably the REE) is present on the surfaces of clay and oxide minerals.

The rainfall on the Poços de Caldas plateau averages 170 n/yr, about 80% of it occurring during a four-month rainy season. The groundwater level is a subdued replica of the surface topography, with recharge from precipitation and discharge into a network of seepages at or near the stream level. The highest point of the water table during the 1981–82 rainy season was about 75 m below the surface of the ridge, or about 75 m above the base of the hill. During the 1981–82 rainy season, in which the rainfall was close to average, the groundwater level in the vicinity of the ore body fluctuated by about 2 m.

Groundwater flow is uniform through much of the deeply weathered ore body, except that near the magnetite dikes it may be controlled by fractures. Permeability measurements in and near the ore body give hydraulic conductivities generally in the range of 10^{-4} to 10^{-5} cm/sec and average groundwater velocities of 1 cm/day (IPT 1982).

Appropriateness of the Selected Analogues

Thorium as an Analogue for Plutonium

The chemical similarity between Pu^{4+} and Th^{4+} has been noted by others (Bondietti and Tamura 1980). The outstanding difference between the two elements is that Pu may exist in several oxidation states, whereas Th is restricted to the 4+ valence.

It is to be expected that Pu compounds would exhibit greater solubility than Th compounds under extreme oxidizing or reducing conditions, but that in the moderate range of Eh and pH that characterizes most natural environments, especially those that would be acceptable for a repository, the behavior would be similar. It is difficult to evaluate the degree of similarity in detail because thermodynamic data for the two elements are incomplete and in some cases of uncertain validity.

The two elements in solution should be limited by the solubility of the oxides ThO_2 and PuO_2 in the absence of complexing agents other than OH^-. The principal Th hydroxy complex at pH levels over 5 is the neutral $Th(OH)_4$ and, between 3 and 5, $Th(OH)_2^{2+}$ (Langmuir and Herman 1980). The equilibrium concentrations for crystalline ThO_2 in contact with pure water are 10^{-14} M at pH levels over 4.5 and rising to only about 10^{-10} M at a pH as low as 3. These concentrations can be as much as five orders of magnitude higher in equilibrium with amorphous oxide, or if complexes are formed with such ions as F^-, SO_4^{2-}, HPO_4^{2-}, or organic compounds. Thus, the theoretical concentrations of Th in dilute natural solutions (pH range 4–9) should normally be below 10^{-9} M (about 0.2 μg/liter), but in some circumstances can be well above this figure. Actual measured concentrations may differ widely from the theoretical equilibrium values not only because of complexing but also because of slow rates of solution and the presence of colloids.

The situation is more complicated for Pu because of oxidation-reduction reactions, but recent estimates of the solubility of crystalline and amorphous PuO_2 (Eisenbud and others 1982, based on Allard and others 1980; Rai and others 1980a; Rai and others 1980b; Jakubick 1979; Jensen 1980) show that under most Eh-pH conditions to be expected in nature the solubilities are closely similar to those for Th. Only under extremely oxidizing alkaline conditions or extremely reducing acid conditions, such as are not likely in a repository environment, would Pu be markedly more soluble.

Solubilities calculated from thermochemical data for the oxides in pure water have only limited applicability to natural environments because of the frequent presence of complexing

agents besides OH^-. The relative stabilities of complexes formed with commonly occurring ligands are an important consideration when judging the chemical similarities of Th^{4+} and Pu^{4+}. Available data are far from satisfactory, but recent compilations (Langmuir and Herman 1980; Cleveland 1979; Lemire and Tremaine 1980; Moskvin and Poznyakov 1979; Phillips 1982) make possible a rough comparison (Table 1). Numbers in the table are logarithms of equilibrium constants for the reactions

$$M^{4+} + L^{n-} = ML^{4-n} (k_1)$$

and

$$M^{4+} + 2L^{n-} = ML^{4-2n} (k_2).$$

Despite the uncertain quality of some of the data, the similarity in constants for the two elements is striking. Where differences exist, they are in the direction that would be expected theoretically—slightly more stable complexes for Pu^{4+}, because its ionic radius (0.98 Å) is a little smaller than that of Th (1.07 Å). Especially notable in the table are the high values for the two organic complexes, reflecting the fact that the solubilities of both elements are markedly increased by the presence of organic materials.*

Neodymium or Lanthanum as Analogues for Curium and Americium

It was recognized early during World War II that the chemical properties of the REE would be similar to those of the transuranic actinides, and the REE were accordingly used to study the chemical properties of the then-scarce artificially produced elements (Seaborg 1958). More recently, Weimer and others (1980) have called attention to the fact that Nd should be an appropriate analogue for Am and Cm. It is to be expected that, like Nd, these actinide elements will exist in the 3+ oxidation state under environmental conditions. The ionic radii are identical for Am^{3+} and Nd^{3+} (1.05 Å), and nearly identical to Cm^{3+} (1.06 Å). Experimental evidence for the analogous behavior of these elements has been provided by laboratory and field experiments. The distribution coefficient (K_d) for Nd in aqueous soil suspensions was found to be nearly identical to those for Am and Cm over a wide range of pH and contact times. Uptake factors for plants grown under laboratory conditions were also found to be similar (Weimer and others 1980).

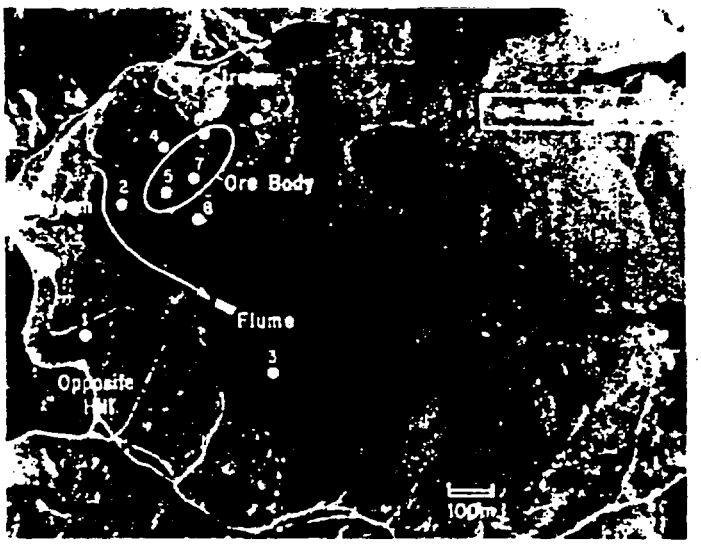

Figure 2. Aerial view of the Morro do Ferro and immediate environs showing the main features discussed in the text.

Although similar laboratory comparisons have not been made with La, its chemical properties are known to be similar to those of Nd. We have found close correlations between the concentrations of Nd and La in various samples of soils and water from the MF environs. Because La is present in higher concentrations in the MF environs, and is more easily analyzed by the method to be described, we have chosen La as the analogue of choice for Cm and Am.

General Investigative Approach

The main objectives of our studies are: (1) to estimate the annual mobilization rate, i.e., the fraction of each substance of interest that is removed from the ore body by either erosion or solubilization; and (2) to understand the underlying hydrologic and geochemical influences that determine the mobilization rate.

Most of the drainage at the MF is down its south face into a small stream (South Stream) that rises from the southwestern edge of the hill (Fig. 2). The flow enters the stream mainly via a series of gullies that receive both surface runoff and groundwater seepage. A flume has been constructed in the location shown in Fig. 2, and instruments have been installed that

*Additional data that appeared while this paper was in press suggest that under some conditions the analogy may not be complete. Allard (1983) reported a stability constant for $PuCO_3^+$ that makes this ion the dominant one in equilibrium with solid PuO_2 at pH's below 8 in slightly reducing solutions with high concentrations of CO_3^{2-} (log CO_3^{2-} (molar) = 0.76 pH − 10.83, higher than the concentrations in most deep groundwater). The stability of this complex, or other carbonate complexes of Pu^{3+}, is corroborated by measurements (Cleveland and others, 1983) of relative concentrations of plutonium in different oxidation states in the very alkaline (pH 10) and carbonate-rich water of Mono Lake. Under conditions to be expected at most repository sites, however, Allard's calculated solubilities for PuO_2 remain below 10 ppb.

permit continuous flow measurements and water sampling at a rate proportional to stream flow. The watershed that drains through the flume has an area of only 0.5 km^2, and there are no impoundments. Surface runoff is thus rapid during a rainfall, and is often accompanied by intense scouring. In a typical shower, the flow in the South Stream can increase in a matter of minutes from a year-round dry weather baseflow of 0.7 m^3/min (stage height 3 cm) to flows as high as 66 m^3/min (stage height 50 cm). In the absence of additional rain, the flow then returns to its baseline value in less than one day. During periods of baseflow, water reaches the stream by seeping from the ground into small tributaries and thus contains thorium that has been mobilized chemically. Hence the stream water contains Th both in dissolved form and in suspended solids that were eroded during previous brief periods of surface flow. During and immediately following a rainfall, the stream water is largely the accumulation of surface runoff.

A second stream (North Stream) rises from the extreme western edge of the north face but receives less drainage from the ore body than the South Stream. Based on Th and flow volume measurements made during the 1981–82 rainy season, it is estimated that mobilization via the north face of the hill is about 25% of that via the south face.

Nine wells, the locations of which are shown in Fig. 2, have been drilled to permit study of groundwater characteristics, and meteorological station at the summit of the hill provides a continuous record of rainfall as well as temperature and wind direction and velocity.

A complicating feature is that water passing through the flume has drained not only the MF, but also the hills that rise from the south bank of the stream (opposite hills). The concentrations of Th and REE in the soils and rocks in this portion of the small drainage basin are much less than in the MF, but are a significant addition to the material mobilized from the ore body. An attempt to correct for this contribution has been made in the following way, and is illustrated only for Th.

It is assumed that the amount of thorium mobilized in particulate form is proportional to the concentration of thorium in the surface soil and the area presented to falling rain. We estimate the average Th concentration in soil over the ore body to be 5,660 $\mu g/g$ and in the remaining portion of the drainage basin to be 180 $\mu g/g$. The area of the ore body is estimated to be 6.2×10^4 m^2. The area of the remaining portion of the drainage basin above the flume is estimated to be 5.0×10^5 m^2. The ore body thus occupies an area that is about 11% of the area of the drainage basin above the flume. With these assumptions, we calculate that the surface of the ore body contributes 80% of the total mobilized particulate thorium.

The relative contributions of the ore body and surrounding areas to the soluble fraction of the thorium flux were similarly estimated, assuming that the contributions would be proportional to the quantity of Th contained in the geologic media through which the groundwater passes. The ore body is estimated to have a volume of 6.2×10^6 m^3 at an average Th concentration of 1,900 $\mu g/g$. The volume of the drainage basin upstream of the flume, but not including the ore body, is estimated to be 1.1×10^7 m^3 with an average concentration of 180 $\mu g/g$. In this way, we have estimated that the ore body contributes 86% of the Th contained in the filtrate. We recognize that this is at best a coarse estimate: for example, the groundwater could be channeled in passing through the magnetite stockwork associated with the deposit, or the concentration of thorium could be solubility-limited and, therefore, independent of the higher concentration of Th in the ore body. Moreover, thorium is undoubtedly dissolved in surface runoff. However, at this stage of the investigation, we do not have sufficient information to permit more than this first approximation.

Methods of Sampling and Analysis

Because of the low concentrations of Th and REE, 20-liter samples are taken, usually in duplicate. During the past year, periodic grab samples have also been taken from the "North Stream," which drains the north slope of the hill; streams draining into the opposite bank of the South Stream; and a stream outside of the plateau, in the town of São João da Bôa Vista.

Eh and pH are measured in the field before the samples are taken to a nearby laboratory where they are filtered through 0.45-μm membrane filters (Millipore). The filtrates are acidified and reduced by evaporation on hot plates from about 20 liter to 0.25 liter to facilitate shipment to New York, where they are analyzed for REE by inductively coupled plasma spectrometry (ICPS), and radiochemically for ^{232}Th, ^{228}Th, and ^{230}Th.

The analytical procedure for Th in water samples and soils has been modified from that of Sill and others (1974; Sill and Williams 1981). In brief, the sample is dissolved in a potassium fluoride–sodium sulfate fusion. This cake is then dissolved in dilute HCl, and the Th separated by co-precipitation on $BaSO_4$, which is filtered and dissolved in $HClO_4$. The Th is purified by extraction into Aliquat-336 (General Mills), washed with 8 M HNO_3, and stripped from the Aliquat with 10 M HCl. The HCl solution is boiled to dryness, and any aliquat and other organics carried over in the extraction are wet-ashed with perchloric-nitric acid. The Th is co-precipitated on cerium fluoride from a 1 M HCl solution of the residue and mounted on a Tuffryn (Gelman) filter for alpha

Table 2. Concentrations of thorium in baseflow water samples compared with two control sites.

Locations	No. Samples	Mean Th concentrations (μg/l)			
			Suspended solids		
		Filtrate[b]	As reported	Normalized[c]	Total
South Stream	55[a]	0.053 ± 0.013	0.53 ± 0.11	0.12	0.59 ± 0.11
North Stream	14	0.027 ± 0.006	0.26 ± 0.03	0.11	0.29 ± 0.04
Opposite hills	13	0.041 ± 0.012	0.64 ± 0.25	0.056	0.68 ± 0.25
São João do Bôa Vista	5	0.011 ± 0.004	0.72 ± 0.33	0.015	0.73 ± 0.33
Hudson River	5	0.005 ± 0.001	0.17 ± 0.10	0.009	0.18 ± 0.10

[a]The South Stream samples are weekly composites, collected by continuous proportional sampler and by grab sampling. All others are grab samples.

[b]Dissolved Th is present chiefly in the form of complexes with fluorine and organic material.

[c]See text for normalization procedure.

spectrometry. The chemical extraction for Th is traced internally by the alpha-emitting ^{229}Th.

The lower limit of detection of ^{232}Th at the 95% confidence level is about 0.12 μg, or about 0.006 μg/liter for a 20-liter sample.

Samples of rock, soil, and suspended solids have also been analyzed by the above procedure or, when a sufficient sample is available, by X-ray fluorescence.

Rare earths in samples of soils, rocks, and suspended solids filtered from our water sample are determined by XRF analysis (Laurer and others 1982). For samples of water and suspended solids, we have adopted the method of Crock and Lichte (1982), in which the Th and REE are separated from the matrix by calcium oxalate precipitation following lithium borate fusion. The REE are separated from Th by sequential elution from a cation exchange resin with 8 M HNO_3 and analyzed by ICPS.

Annual Thorium and Lanthanum Mobilization Rates

Thorium

Our estimates of the annual Th flux cover the period May 1981 through April 1982, during which time the rainfall was 177 cm (similar to the 30-year annual average of 170 cm). Data are available for both the suspended solids and the filtrates.

The flow in the South Stream during much of the year is constant, at 54 m^3/hr. This "baseflow" accounts for about 85% of the year-round drainage via the South Stream. The results of Th analysis of samples of the baseflow are given in Table 2, together with analyses of water samples from other locations for reference.

Th carried in suspended particulates ranges between 91% and 99% of the total Th flux for all localities, with the weighted mean being 92%. The flux of particulate Th is highly dependent on the concentration of total suspended solids. The suspended particulate data in Table 2 are given both as μg Th/liter and normalized as μg Th/liter per milligram of suspended solids per liter. The mean concentration of thorium in the filtrate samples varies only from 0.011 μg/liter at São João da Bôa Vista to 0.053 μg/liter in the South Stream.

From the data of Table 2, and our knowledge of annual dry weather flow (4.75×10^5 m^3), we conclude that the annual flux of Th mobilized by the baseflow, including both the North and South Streams, is 0.027 kg in soluble form and 0.26 kg in suspended particulates. We assume the filtrate component during periods of baseflow results from solubilization of Th by groundwater. Based on our estimate that the ore body contains 30×10^6 kg of Th, groundwater mobilization under baseflow conditions is at a rate of $0.027/(30 \times 10^6) = 9.1 \times 10^{-10}$ yr^{-1}. Mobilization of Th in particulate form during periods of baseflow is similarly calculated to be 8.5×10^{-9} yr^{-1}. The total mobilization during dry (baseflow) periods is thus 9.4×10^{-9} yr^{-1}. It is reasonable to assume that much of this fraction is due to resuspension and bank wash of stream sediments.

Fifty-four samples were collected during ten periods of rainfall in the 1981–82 rainy season. The Th flux (ϕ_{Th}) in particulates was found to be highly correlated ($r = 0.9$) with stream flow (Q) according to the following equation:

$$\phi_{Th} = 5.4\, Q^{2.2}$$

where: ϕ_{Th} = mg Th/min and Q = m^3/min

Twenty-five filtrates of the 54 storm samples were also analyzed, but no relationship between Th concentration and flow was evident. The mean filtrate concentration, 0.22 ± 0.06 μg/liter, was assumed to be representative of stormflow filtrates. This value is four times greater than the average concentration (0.05 ± 0.01 μg/liter) found in the South Stream baseflow filtrates. This may be due to the presence of a greater

quantity of fine particulates (<0.45 μm) in the storm water filtrates.

The annual Th flux during periods of rainfall is estimated to be 0.015 kg in "soluble" form, and 27.2 kg as suspended solids. Not surprisingly, 99% of the annual flux of Th particulates is mobilized during periods of rainfall. Mobilization of Th in particulate form is at an annual rate of $27.2/(30 \times 10^6) = 9.1 \times 10^{-7}\ yr^{-1}$. The mobilization rate of soluble Th by stormflow is $5.0 \times 10^{-10}\ yr^{-1}$. The total annual mobilization rate in soluble form is thus $5.0 \times 10^{-10} + 9.1 \times 10^{-10} = 1.4 \times 10^{-9}$.

Lanthanum

There are no previously published estimates of the quantity of La in the ore body, but this can be approximated from our analysis of 101 samples obtained from the well cuttings and exploratory trenches and tunnels. The mean ratio of La to Th in the individual samples is 2.3 ± 0.4. Applying this ratio to the quantity of Th in the ore body provides an estimate, 2.3 × 30,000 tonnes = 69,000 tonnes, of La.

We have previously estimated that there is about 4,000 tonnes of Th upstream of the flume, but outside of the ore body. The observed La/Th ratio in this region for 71 samples is 7.6 ± 0.5, which permits an estimate of 30,400 tonnes of La in this portion of the drainage basin.

The observed La/Th ratio in 12 samples of suspended solids taken during storm flow is 2.9 ± 0.2, similar to the ratio found in the ore body. From this, and knowing the annual Th flux, we estimate that the quantity of La mobilized in particulate form is 2.9 × 27.2 = 78.9 kg. The annual mobilization rate of La in particulate form is, therefore, 0.079 tonnes per year/69,000 tonnes = $11 \times 10^{-7}\ yr^{-1}$, which is in close agreement to the $9.1 \times 10^{-7}\ yr^{-1}$ estimated to be the mobilization rate for Th in particulate form.

No samples are available from which the rate of stormflow solubilization of La mass can be determined. However, 20 samples from the baseflow regime have been analyzed, yielding a mean of 0.28 ± 0.03 μg/liter, which gives an estimated mobilization rate by groundwater solubilization of 1.6×10^{-9} per year, which is similar to the rate for Th (9.1×10^{-10} per year).

Significance of These Findings

The mobilization rates have been estimated above for two mechanisms, surface erosion by the action of rainfall and solubilization by groundwater. Groundwater intrusion is more relevant to a geological repository, although the possibility of eventual exposure of a deep rock repository due to erosion cannot be excluded.

Table 3. Use of the Th analogue to estimate the concentration of Pu^{+4} in South Stream water.[a]

A.	Mass of Th in MF ore body	= 30,000 tonnes
B.	Average concentration of Th (<0.45 μm) mobilized by groundwater	=0.053 μg/l
C.	Expected accumulation of ^{239}Pu in U.S. by 2050 (U.S. Dept. of Energy, 1980)	= 2,000 tonnes
D.	Predicted $^{239}Pu^{+4}$ concentration in South Stream water[a]	= CB/A = 0.0035 μg/l
E.	Maximum allowable concentration of ^{239}Pu in waste water (Code of Federal Regulation)	= 0.08 μg/l

[a]Based on assumptions given in text.

The mobilization rates of the natural analogues of Pu, Am, and Cm have been found to be so low that despite the long half-lives of some of the actinide isotopes, essentially complete decay would take place in situ if the analogues we have studied are valid. For example, if we assume that the annual groundwater mobilization rate of ^{239}Pu will remain constant at the present rate for Th, the mean life of the deposit would be 7×10^8 yr compared to 3.4×10^4 yr due to radioactive decay of ^{239}Pu.

It is all the more remarkable that the mobilization rates are so low in view of the characteristics of the ore body and its environs. Site selection criteria for geological repositories (NRC 1978) require that they be placed at great depths in unweathered rock, under hydrological conditions that would assure long transit times for nuclides carried by groundwater. The MF ore body meets none of these requirements—depth of burial, condition of the host rock, or transit time of groundwater to potable water. Moreover, preliminary results of speciation studies conducted by Miekeley and others (1982) suggest that a major fraction of the Th in groundwater filtrates is complexed with organic acids which presumably form from the decomposition products of vegetation that percolate to the groundwater through the porous rock. Nevertheless, the mobilization rate for Th is so low that if the Th-Pu analogy is valid, the South Stream filtrates would meet the presently accepted waste discharge standards if all the Pu expected to exist in the United States by 2050 (U.S. DOE 1980) were to be emplaced under conditions comparable to those existing at the MF. We come to this conclusion in the manner shown in Table 3, in which it is estimated that under such conditions the Pu concentration would be about 5% of the maximum permissible concentration (CFR, Title 10) allowed by the regulations of the Nuclear Regulatory Commission.

This conclusion would be particularly valid if the actinide wastes were not exposed so directly to atmospheric oxidation as

is the MF deposit, since this exposure might lead to a change in the oxidation state of plutonium, and hence to greater mobility. Measured Eh (+400 to +700) and pH (4.4 to 7.1) in soil and water at MF. despite exposure to air, are within the field where plutonium has the same valence as thorium and should be similarly immobile (Eisenbud and others 1982). Nevertheless, access to air always offers the possibility of oxidation of some of the plutonium. For a repository well sited and deep underground, such access of air by deep erosion is extremely unlikely. A more probable scenario for repository breaching would be invasion by groundwater, which implies no extremes of either Eh or pH, and in this case the behavior of plutonium should closely mimic that of its analogue.

If neptunium is also present in the waste, the need to site a repository so as to minimize the chance of exposure by erosion is still more urgent, because neptunium is more easily oxidized than plutonium, and because its half-life is longer (2.1×10^6 yr for ^{237}Np). Quadrivalent neptunium is not as close a chemical analogue of thorium as is plutonium, but still should be similarly immobile as long as conditions remain reducing. Thus, the data on thorium migration at MF indicate clearly that none of the transuranic elements would be appreciably mobilized by groundwater contact with the waste in a repository.

Our finding that these analogues of the transuranic actinide elements are so extraordinarily immobile is consistent with findings at the site of the natural reactor in the Oklo uranium mine in Gabon, West Africa (Cowan 1976; Curtis and others in preparation). It is known that about 1.8 billion years ago, conditions in this uranium deposit were such as to sustain criticality for an estimated 10^4 to 10^5 years. The transuranic elements produced by this lengthy episode have long since decayed, but the stable isotopes of the end products of decay have remained, and mass spectrographic studies of rock samples collected in the vicinity of the ore body have shown that minimal migration of actinide elements occurred.

The investigations at Oklo and MF are not yet complete, but the data obtained thus far from both studies suggest that, although the actinide elements are hazardous when absorbed into the body, there are geochemical barriers that would, under ordinary conditions, block these elements from leaving a waste repository and being incorporated into food and water.

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Section 3 - Dose and Cumulative Curie Release

Disposing of High-Level Radioactive Waste – The U.S. Regulatory Approach

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Abstract

This paper focuses on the regulatory approach being used in the United States to manage the disposal of high-level nuclear waste. A summary of the roles of the various governmental agencies in the United States that hold responsibility in these areas is provided. The paper then describes in general detail the regulatory requirements being applied in other countries and provides a frame of reference for a detailed discussion of the current U.S. regulations, including a review of areas of uncertainty. Uncertainties and perceived difficulties that have been identified in the proposed high-level radioactive waste environmental standards and related implementing regulations in the United States are discussed. Modifications to the current U.S. regulations are suggested that might provide a level of protection in the United States that is consistent with that being used internationally. Finally, some thoughts on developing a scheme of apportioned risk to the public from nuclear sources are posed for consideration and discussion.

Introduction

A current topic of discussion among experts in the U.S. high-level waste disposal field is the regulatory framework in place in the United States. Many agencies, including the Nuclear Waste Technical Review Board (the Board), have expressed their concerns about existing standards and regulations. In its *First Report* (NWTRB, March 1990), the Board stated that although licensing standards and criteria must be adequately conservative on the side of safety, they should not, for example, foreclose at the outset a candidate site that subsequently could be proved suitable based on sound scientific considerations. In its *Second Report* (NWTRB, November 1990), the Board expressed its belief that the current regulatory framework could be improved. It is the view of the authors that the existing regulatory framework *could* foreclose a site at the outset and that the framework *should* be improved.

This conference has provided the authors the opportunity to review the U.S. regulatory framework within the international context. Although members of the Board and staff, the authors would like to state that the thoughts presented do not necessarily represent the views of other Board members, nor of the Board as a whole. Especially the suggestions for making changes to the regulations and the concept of apportionment of radiation risk should be considered the personal opinions of the authors.

The Hierarchy of the U.S. Nuclear Waste Management Program

In the United States, the responsibility for disposing of high-level nuclear waste has been divided among a variety of federal agencies. The U.S. Department of Energy (DOE), through its Office of Civilian Radioactive Waste Management (OCRWM), is responsible for developing and operating a subsurface geologic repository. In 40 CFR 191, the Environmental Protection Agency (EPA) sets the radiation protection standards and public exposure limits that must be met by the repository during the operational and postclosure periods (EPA 1985). In 1987, a federal appeals court remanded a portion of 40 CFR 191 to the EPA for revision, where it is still under discussion. The Nuclear Regulatory Commission (NRC), which is responsible for implementing the EPA standards, sets criteria for doing so in 10 CFR 60 (NRC 1983). The NRC is the body that will ultimately issue a license for the operation of the repository and for the disposal of high-level waste. In addition, the DOE has set site-selection criteria, which are codified in 10 CFR 960 (DOE 1984).

Each of the above groups is reviewed by one or more entities that advise and comment upon their technical efforts. The EPA's regulations are reviewed by its Science Advisory Board, which, upon request of the EPA or on its own initiative, examines pending EPA standards and rules. The NRC has had two such advisory bodies. The Advisory Committee on Reactor Safeguards had oversight responsibility in the area of waste management of both high- and low-level waste. These tasks were assumed in 1988 by the Advisory Committee on Nuclear Waste. The DOE and OCRWM receive advice and oversight from, among others, the Nuclear Waste Technical Review Board, which is the only group appointed by the President that provides Congress with regular reports on the technical and scientific aspects of the DOE's program.

Survey of Non-U.S. Nuclear Waste Guidelines and Regulations

In 1985, the International Commission on Radiation Protection (ICRP) published ICRP Publication 46, *Radiation Protection Principles for the Disposal of Solid Radioactive Waste* (ICRP 1985). It proposed a dose limit to the public of 1 mSv per year, excluding medical and natural sources, and recommended apportionment of radiation risk for other sources by national authorities. The International Atomic Energy Agency (IAEA) issued reports 96 (1989a) and 99 (1989b) in its Safety Series. These documents presented criteria and guidance for the underground disposal of nuclear wastes, and were directed to the disposal of high-level waste, such as spent fuel.

Individual governments have used these documents in the development of their own standards and regulations.[1] In 1987, the International Energy Associates Limited prepared and published a compilation of the various nuclear waste regulations by country (IEAL 1987). In Table 1, which is largely drawn from that document, the term "protection limit" is the maximum allowable exposure limit for members of the public during the operational phase of repository activities. The repository "system performance goal" is the maximum dose to a member of the public after closure. It is clear that the values set bracket the value

[1] *The terms "standard" and "regulation" are sometimes used interchangeably. For the purpose of this paper, the term "standard" connotes the base or fundamental level of performance or protection. "Regulation" and "rule" refer to specific criteria by which a standard is to be met.*

TABLE 1

Comparison of National Nuclear Waste Regulations

Country	Arg[2]	Belgium	Canada	FRG	France	Japan	Nordic	Switzerland	U.K.	U.S.
Protection Limit (preclosure) (mSv/yr)	0.3	5	5	0.3	none	0.5	0.1	5	0.1	0.25
Repository System Performance Goal (mSv/yr)	0.3	none set	geologic criteria set	0.3	none set	0.05	0.1 plus activity inflow (proposed)	0.1	0.1	0.25 plus curie release

of 1 mSv annually, originally proposed by the ICRP as the principal limit for members of the public. It also should be noted that, of the several nations that have set performance limits, only the United States has chosen to set activity release limits on repository performance in addition to dose limits. It is difficult to precisely compare the curie release limit with a dose limit. Alternatively, in the U.S. approach, if one uses the excess health effects goal of one health effect every 10 years, which is the basic risk goal of the EPA Standard, to calculate dose, a value of approximately 5×10^{-5} mSv/yr is obtained. This value is well below conventional levels of detection, or measurement, and is 1/5000 of the EPA's 1,000-year, individual protection requirement of 0.25 mSv/yr. Alternatively, the proposal by the radiation protection and nuclear safety authorities in Denmark, Finland, Ireland, Norway, and Sweden (Nordic) contains both a individual protection limit and limit on activity inflow. While not yet in force, the limitation on activity inflow is expected to restrict the average population dose to approximately 0.1 mSv annually.

A further point of comparison is the relative level of protection of the public during the pre- and postclosure phases of repository operation. The U.S. regulations require a very large reduction in risk upon closure; the criteria set by Germany and the United Kingdom, for example, make no such increased demands on the performance of the repository after closure. Those specified or under consideration in Switzerland and Japan call for intermediate reductions in risk upon disposal.

The unique element that distinguishes the U.S. approach from those in other countries is that, unlike most countries, the United States has predefined or specified "subsystem performance criteria" before the proposed site has been investigated, or the repository designed.

Description of U.S. High-Level Waste Regulations

The development and operational activities that support the proposed high-level radioactive waste repository are controlled by (1) the EPA Standard 40 CFR Part 191, *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High Level and Transuranic Radioactive Waste*; (2) the NRC Regulation 10 CFR Part 60, *Disposal of High-Level Radioactive Waste in Geologic*

[2] *Private Communication - Dr. D. Beninson, ICRP, Argentina.*

Repositories; and (3) the DOE Regulation 10 CFR Part 960, *General Guidelines for the Recommendation of Sites for Nuclear Waste Repositories.* The DOE regulations provide guidance in determining the suitability of a site, whereas the NRC regulations define the detailed requirements placed on both the host rock and the engineered features of a repository. Both of these regulations are subordinate to the EPA Standard 40 CFR 191, which defines the absolute level of protection that a repository must provide to the public.

During the NRC hearings on Part 60 before the Nuclear Regulatory Commissioners in 1982, the director of the Waste Management Division stated that merely meeting the requirements included in the–then proposed–version of Part 60 would not assure compliance with Part 191 (then in draft form). This statement was reaffirmed by the NRC staff in December 1990, when, in a presentation to the Advisory Committee on Nuclear Waste, the director of the NRC's Office of Nuclear Materials Safety and Safeguards stated that the specific subsystem requirements of Part 60 were neither necessary nor sufficient in fulfilling the requirements of Part 191.

It also is important to understand that when the regulations and the standards were being developed, all repository sites then under consideration were in saturated media. This changed in 1983 when the proposed repository depth was reduced for the site at Yucca Mountain, Nevada. However, the rules were not amended at that time to reflect the conditions at Yucca Mountain. The staffs continued to assume (1) that the site would be saturated and (2) that borehole emplacement of the waste packages would be used. In 1986, however, Part 60 was amended to explicitly consider emplacement in the unsaturated zone.

Analysis of the Standard and Regulations

Below is a discussion of the repository subsystems, the containment and performance requirements, and the regulations. Both Parts 191 and 60 provide definitions of the boundaries, real and imaginary, that constitute the repository system. Pertinent definitions from both parts are represented graphically in Figures 1 and 2. Also illustrated in Figures 1 and 2 are the performance requirements applicable at the boundaries, which are discussed separately below.

Repository subsystems

The *repository* consists of several areal or volumetric entities. The innermost unit is the waste package, which includes the waste form (either spent fuel or vitrified wastes), its container (canister and overpack), and other materials within or immediately surrounding the container. Normal usage and specifications by the NRC staff limit the extent of the waste package to the interior surface of its emplacement site. The *waste package* is a component of the engineered barrier system (EBS). The EBS also includes the *underground facility*, which comprises all excavated areas and their associated backfilling materials.

The *controlled area* is located beyond the EBS. The two regulations define different starting points for the controlled area. In addition, the horizontal extent of the controlled area is defined somewhat differently in each regulation. These differences are noted in Figure 1.

The *disturbed zone* is defined as that portion of the controlled area whose physical and/or chemical properties will be changed by the heat released by the emplaced wastes or as a result of the construction of the underground facility.

Containment and performance requirements

The criteria in Parts 191 and 60 place requirements on the performance of the overall repository system and some of its components. (See Figures 1 and 2.)

The overall criterion for repository performance requires that there be only one chance in ten that the total releases to the accessible environment over 10,000 years exceed the quantities listed in Table 1, Appendix B of Part 191, under expected operating conditions (Section 191.113). Two other NRC mandated performance criteria apply at the waste package and EBS boundaries. The respective requirements are that "substantially complete containment" within the waste package be maintained for a minimum of 300 years and that the annual release of certain radionuclides from the EBS be less than 10^{-5} of a portion of the radionuclide inventory at 1,000 years.

Analysis of the regulations

Part 191. Thus far, little attention has been paid to Subpart A of Part 191. That portion of the standard is similar to the requirements of 40 CFR Part 190 (EPA 1977), which is the EPA protection standard for nuclear processing facilities, such as reprocessing plants. Part 190 and Subparts A and B of Part 191 all have dual standards of both dose and release limits. Subpart A of Part 191, which was not remanded, is now in effect and limits the annual exposure to any member of the public to 0.25 mSv/yr to the whole body and to 0.75 mSv/yr to any critical organ during storage and other waste handling operations. Although this exposure limit is lower than the value of 1 mSv previously allowed by the NRC in Section 20.105 for reprocessing operations, it is similar to limits set by other countries; for example, in Germany the limit is 0.30 mSv/yr. After closure, the dual limits of Subpart B would apply, rather than those of Subpart A.

The postclosure curie limits were chosen by calculating the individual doses received by the public from an ore body containing an amount of uranium comparable to the original uranium in the spent fuel and estimating the resultant number of excess health effects. It was from this calculation that the base health effect limit of 1,000 health effects in 10,000 years was obtained. The curie release limit has been estimated to have the capability of delivering a maximum annual dose of perhaps 5×10^{-5} mSv. This represents a reduction in the allowable dose of approximately 5×10^{3} after the repository closes. The reason cited by the EPA staff for requiring this massive reduction in allowable dose is that they believed that the uncertainties of geologic data and performance assessment calculations required that very low allowable releases be mandated. Thus, the EPA has attempted to accommodate for its perception of the uncertainty inherent in site-characterization studies and performance assessments by setting very low limits and using probabilistic requirements.

Commenters on Part 191 have noted the stringency of the level of releases noted above. In addition, questions have been raised as to the toxicity of certain of the radionuclides, such as carbon-14, and whether or not such elements deserve to be included in Table 1 at all. In the case of carbon-14, it turns out that (1) during reprocessing all the carbon-14 contained in spent fuel is available for release, (2) no limit has been placed on the release of carbon-14 from operating reactors, and (3) the Subpart B limit

Figure 1
Regulatory Boundaries — Outside of Repository

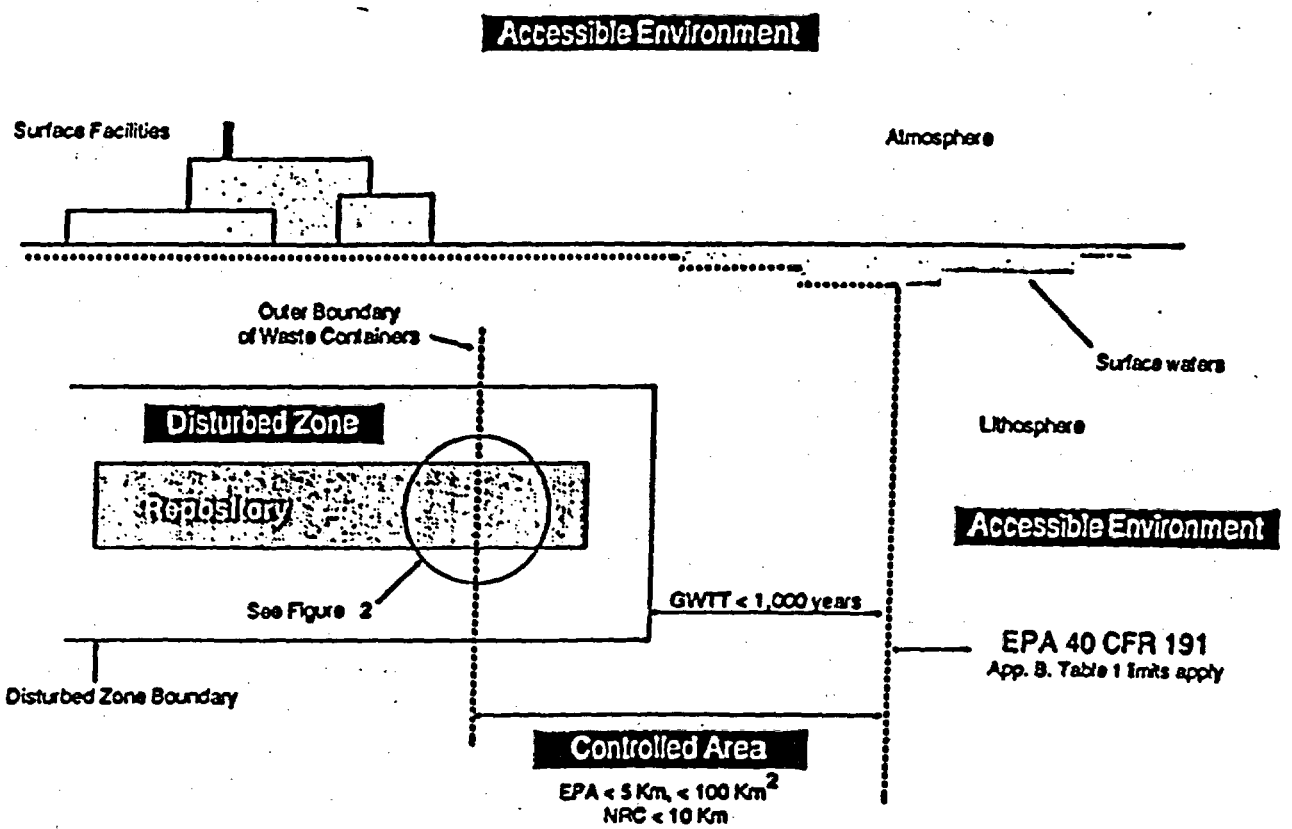
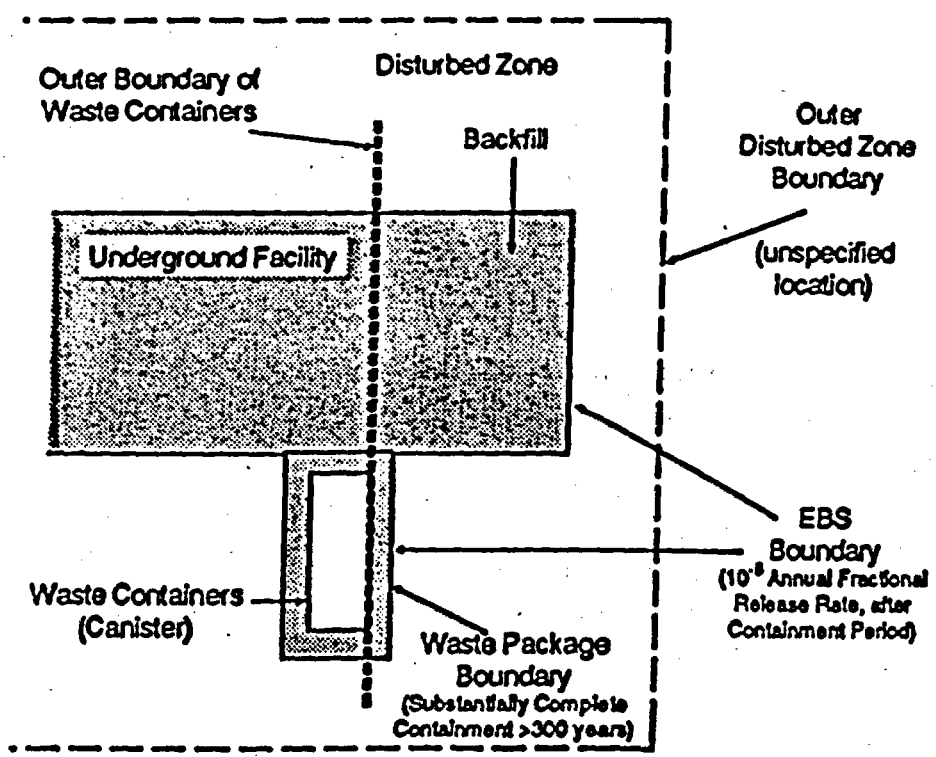


Figure 2
Regulatory Boundaries



100

of Part 191 is two or three orders of magnitude below what is cosmogenically produced annually, thereby rendering moot the very low limit on the release of carbon-14.

Part 60. The NRC staff adopted the approach of assuming a multiple-barrier system and setting individual performance requirements on the barriers, or successive concentric zones, that comprise the repository. Thus, there is a containment requirement on the interior subsystem (the waste package), an annual fractional release limit on the next layer (the EBS), and a limit on the groundwater travel time (GWTT) to the accessible environment from the disturbed zone.

1. Containment Requirement. The NRC staff recognized the impracticality of setting corrosion limits on the waste form (i.e., spent fuel and vitrified waste) since the chemical composition and physical form of the waste materials and consequently their corrosion behavior will be determined by fabrication and service conditions beyond the control of the DOE. As a consequence, the staff developed qualitative rather than quantitative criteria on containment. In particular, the phrase "substantially complete containment" provides no quantitative criteria that would allow the licensee to estimate either the permissible number of failed canisters or the quantity of material dissolved into the host rock. Nor does Part 60 provide any definition of what "containment" means. For example, members of the NRC staff have stated that any loss of integrity of a waste package, even a pinhole penetration, would constitute loss of containment, thus contributing to exceeding this particular criterion. In late 1984, the NRC circulated a draft proposal for quantifying the term "containment." Essentially, the proposal was to apply the post-containment annual release limit to the containment period, thus enabling the calculation of the amount of inventory allowed to leave the waste package. However, that approach has been abandoned without public explanation. The NRC now has an effort underway at its Center for Nuclear Waste Regulatory Analysis to provide a further definition of these terms, but this work appears to be directed at a nonquantified specification that will not assist designers in meeting the regulation.

2. Annual Release Limit. Section 60.113(a)(1)(ii)(B) specifies that subsequent to the containment period, only one part in 100,000 of the radionuclide inventory present at 1,000 years in the repository may be released annually. That release limit is further qualified to the extent that only those radionuclides that individually constitute more than 0.1 percent of the inventory need be considered. This limitation also is subject to the requirement that the total releases must not exceed the values set in Table 1, Part 191.

3. Groundwater Travel Time (GWTT). In Part 60, the criterion placed on the geologic setting is that "the geologic repository shall be located so that (the) pre-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment shall be at least 1,000 years . . . " This requirement illustrates the fact that the regulation was written with the presumption that the repository would be located in a saturated zone. It is necessary to note that an amendment to the regulation clarifies the intent of the regulation to be equally applicable to all repositories, whether they are in either saturated or unsaturated geologic media.

Three questions arise about this criterion, particularly for a repository in an unsaturated zone: (a) Since there is free access to the accessible environment through a direct atmospheric pathway, what will be the impact upon meeting this requirement for radionuclide transport when the release of volatile radionuclides is considered? It would appear that the release of both iodine and carbon (as C-14) is likely to occur promptly upon failure of the principal container (b) How should one define and quantify a property such as GWTT when there is not a continuous water phase that would support the transfer of radionuclides in the conventional sense? (c) Given that one may be able to define and measure GWTT, what constitutes the "fastest travel time?" It would be expected that any measuring technique will not

define a single value, but give a range or distribution of values. Consequently, there will be a range of possible values, but no specification as to how the lowest value is to be chosen. For example, one might select a 90-percent cutoff–or a 99 percent.

The concept of specifying that certain characteristics be evident in the host rock or geologic setting is, of course, reasonable and prudent. But it may be equally prudent to include such characteristics as desirable rather than making them requirements or specifying their value before the site has been specified and characterized.

Oversight Comment on the Regulations

There has been an extensive commentary on the standards and the regulations. This has included (1) a report by the Waste Isolation Systems Panel (WISP 1983), (2) the review of the EPA standards by the EPA's Science Advisory Board (SAB 1984), (3) the preparation of an analysis of the then proposed Part 60 by a group of NRC contractors (Chu et al., 1983), (4) concerns voiced by the ACNW in 1989 on Part 191, (5) a report by the National Academy of Sciences in 1990 (NAS 1990) and; (6) recommendations by the NWTRB in 1990 (NWTRB, March and November 1990).

Advisory and oversight groups have supported as well as found fault with both the standard and implementing regulations. However, the general thrust of the comments has been negative. It appears that three principal concerns are (1) the application of very low, or overly stringent, release rates (e.g., carbon-14) as the principal repository performance measure; (2) the use of probabilistic factors to control application of the primary performance limit; (3) the imposition of numerous, ill-defined, and ambiguous restrictions on the design, investigation, construction, and operation of the repository before the site has been identified and characterized, or the repository designed.

Recommendations

Since the EPA is in the midst of a revision of the standard, the authors believe that it would be appropriate to encourage the EPA staff to request the EPA Science Advisory Board to set up a new subcommittee to provide guidance on the format and content of the re-issued rule. We also encourage the development of a national consensus on a "system of permissible risk" from such sources as nuclear reactors, other parts of the nuclear fuel cycle, and waste processing and disposal operations. We would not recommend the inclusion of natural and medical exposures in this system of permissible risk. The magnitude of this permissible risk could be in the range of observed variations in natural background doses, up to 1.0 mSv/yr or a risk level of about 10^{-5}. In fact, such a system seems to be developing in the United States now. The system is based principally on the analysis of radiation risks by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1988) and the Committee on Biological Effects of Ionizing Radiation (BEIR V 1990). Recommendations for a dose limit to the general public of 1 mSv/yr averaged over any five consecutive years were made recently by the International Council on Radiation Protection (ICRP 1990). Similar recommendations had been made earlier by the National Council on Radiation Protection (NCRP 1987) and by the ICRP (ICRP 1985) and

the IAEA (1989b). The UNSCEAR recommendations called for an annual effective dose equivalent not to exceed 1 mSv/yr for continuous (or frequent) exposure from man-made sources other than medical.

The DOE, in Order 5400.5 (DOE 1990), reduced its radiation standards for the general public to 1 mSv average annual effective dose equivalent. The NRC has proposed similar limits and requirements in its draft 10 CFR Part 20, whereas the EPA is in the process of developing similar guidance for all federal agencies.

The next logical step is to apportion parts of this public radiation standard to various categories of radiation exposure (sources). To some extent, this has been done on a de facto basis. For example, the EPA in its uranium fuel cycle standard, 40 CFR 190, used a value of 0.25 mSv/yr. This standard excludes mining, transportation, and waste disposal.

We suggest that the EPA use a radiation standard based on an annual radiation dose in 40 CFR 191 and that a dose 0.1-0.3 mSv/yr to the maximally exposed individual be considered. This represents 10-30 percent of the 1 mSv/yr presently in use and proposed for the general public for all man-made radiation exposure other than medical applications. Such a radiation standard is risk based, taken from a system with which there is broad national and international experience, generally understood by members of the public, defensible in the licensing process, consistent with international practice, and fully protective of the health and safety of the public.

Regardless of the EPA's ultimate decision on revising 40 CFR 191, we believe that the NRC should seriously reexamine, and perhaps revise, 10 CFR 60. The detailed criteria supporting the standard should be directly related to meeting the standard. The regulations should be neither ambiguous nor should they act to direct or limit the design or technical approach used to meet the standards.

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A Study of the Isolation System for Geologic Disposal of Radioactive Wastes

Waste Isolation Systems Panel

BOARD ON RADIOACTIVE WASTE MANAGEMENT

Commission on Physical Sciences, Mathematics, and Resources

National Research Council

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This report has been reviewed by a group other than the authors according to procedures approved by a Report Review Committee consisting of members of the National Academy of Sciences, the National Academy of Engineering, and the Institute of Medicine.

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8

OVERALL PERFORMANCE CRITERION FOR GEOLOGIC WASTE DISPOSAL

8.1. INTRODUCTION AND SUMMARY

In previous chapters we have described and evaluated the state of knowledge of the components of the geologic waste-disposal system. In Chapter 9 we present the calculated performance of the overall system of these components in terms of expected and possible long-term releases of radionuclides in the environment and radiation doses therefrom. To assess whether or not this predicted long-term performance is adequate, an overall performance criterion is needed.

No overall performance criterion for geologic waste disposal has yet been adopted by the federal agencies. The Environmental Protection Agency (EPA) has been working for several years to develop such a standard. The EPA staff has written several internal drafts and supporting technical reports, and a proposed standard has recently been issued for public review. This begins the formal process for promulgating an EPA regulation that would adopt some overall performance criterion as its standard. The Nuclear Regulatory Commission (NRC) has issued both draft and proposed final regulations, which contain detailed numerical criteria for individual components and are intended by NRC to implement the proposed EPA standard.

As is explained later, the panel finds technical flaws in EPA's derivation and justification of its proposed standard. In reviewing NRC's proposed regulations, we conclude that they are premature in that they purport to implement an overall EPA standard not yet issued and not yet subjected to the review process wherein the bases and merits of the standard can be fully examined. We also find flaws in NRC's technical basis for some of its numerical criteria. Similar reservations concerning NRC's numerical criteria have been noted by the National Research Council's Board on Radioactive Waste Management (Wilson and Krauskopf 1981). Consequently, we are reluctant to adopt the EPA and NRC approaches for selecting an overall criterion for this study.

The Department of Energy (DOE) has not adopted, to our knowledge, an interim overall criterion for evaluating waste-isolation performance, although authors of studies by DOE contractors have assumed values of the individual-dose rate as criteria for comparing with calculations of doses from radionuclide migration and release.

Therefore, the panel found it necessary to adopt its own performance criterion for the purposes of this study. We conclude that the most meaningful and useful form of the criterion is the annual or lifetime radiation dose to an individual exposed at some future time to radionuclides released to the environment from a geologic repository. We have adopted as our criterion an annual radiation dose of 10^{-4} Sv* to an individual, averaged over his lifetime, calculated at all future times. This dose criterion is to apply to estimated average annual doses resulting from events that have a high probability of eventually occurring.

When the predicted radiation exposure is due to some unexpected release of radioactivity, i.e., a release probability of less than unity, then the predicted dose should be weighted by the probability before comparing with this performance criterion. This is qualitatively similar to the present NRC-EPA practice of allowing greater individual radiation exposures for suitably improbable accidental releases of radioactivity.

One reason for adopting the individual-dose criterion is that the individual dose from released radionuclides can be expressed as some fraction of the radiation dose that each individual receives from background radiation. Another reason is the considerable precedence for individual-dose criteria in radiation safety, as is reviewed in Section 8.2. In Section 8.3 we discuss why a criterion based on radiation exposures as low as reasonably achievable is not meaningful at this time. In Section 8.4 we discuss the features of an alternate approach, in terms of a criterion based upon population risk or collective population dose. In Section 8.5 we review the EPA proposed population-risk-based standard, and we discuss our reasons for not adopting such a performance criterion or the activity release limits derived therefrom.

In Section 8.6 we review the numerical criteria proposed by the NRC to implement the EPA's proposed standard, and we discuss the reasons for not adopting the NRC staff's numerical criteria for the purpose of this study. In Section 8.7 we discuss the compatibility of geologic isolation with the criterion of suitably low radiation dose to future individuals.

8.2. PRECEDENTS FOR AN INDIVIDUAL-DOSE CRITERION

8.2.1. Considerations in Specifying an Individual-Dose Criterion

Two arguments for an individual-dose criterion are (1) individual dose can be more meaningfully predicted than can a population dose and (2) even if a population-dose criterion is adopted, an individual-dose criterion would necessarily also be included to prevent undue individual exposure.

*1 sievert (Sv) = 100 rem.

A consideration in selecting a value for an individual-dose criterion for geologic disposal is that the value of the individual dose be low enough that the number of individuals potentially exposed is not a major consideration. We examined several existing radiation standards developed for protection of the public from radionuclides released to the environment. Natural background radiation exposure was also considered in selecting a value of the individual dose criterion.

8.2.2. Recommendations by the International Commission on Radiological Protection

The International Commission on Radiological Protection (ICRP) concludes that a lifetime exposure to an individual member of the public of 10^{-3} Sv/yr corresponds to an acceptable level of risk, based on a total stochastic risk factor of about 1.4×10^{-2}/Sv (International Commission on Radiological Protection 1977). The commission recommends a limit of 5×10^{-3} Sv for the exposure to an individual in any one year, and it concludes that this would result in an average dose rate equivalent of less than 5×10^{-4} Sv/yr to members of the public. These exposures are considered to be in addition to exposure from background radiation.

8.2.3. Federal Radiation Council Guidance

In 1960 the U.S. Federal Radiation Council (FRC) issued guidance for exposure to members of the public from federally sponsored or licensed activities. The basic radiation protection guide (RPG) is 5×10^{-3} Sv/yr to any individual (the maximally exposed individual). The FRC also stated that the average annual dose to a particular exposed group should not exceed 1.7×10^{-3} Sv/yr (U.S. Federal Radiation Council 1960). These RPGs are also implied as the basis for most of the present limits in the U.S. NRC (1982a, Appendix B) regulation concerning concentrations of radionuclides in effluents from licensed facilities.

8.2.4. The Environmental Protection Agency's Fuel Cycle Standard

The EPA published an environmental protection standard for the uranium fuel cycle that contains a dose equivalent limit of 2.5×10^{-4} Sv/yr to any individual member of the public (U.S. Environmental Protection Agency 1976b). This standard has been incorporated by the NRC in its regulations for nuclear fuel cycle licensees (U.S. Nuclear Regulatory Commission 1982a) and has been proposed by the NRC for inclusion in its rules on commercial low-level (U.S. Nuclear Regulatory Commission 1981d) and high-level (U.S. Nuclear Regulatory Commission 1981c) waste disposal. In the EPA high-level waste standard, it is proposed to

include the dose limits for application to the above-ground preemplacement operations of a high-level repository (U.S. Environmental Protection Agency 1982).

8.2.5. Other Regulatory Standards

A number of other regulations have adopted radiation dose or dose rate to the individual as the radiation safety criterion, with values varying from below 10^{-4} Sv/yr up to 5×10^{-3} Sv/yr. Some of these regulations were developed for situations wherein the radionuclides are already present in the uncontrolled environment. They include:

- 5×10^{-5} Sv/yr, uncontrolled release of biomedical wastes containing tritium and carbon-14 (U.S. Nuclear Regulatory Commission 1981b)
- 10^{-4} Sv/yr decommissioning of nuclear facilities (U.S. Nuclear Regulatory Commission 1981a)
- 2.5×10^{-4} Sv/yr, equivalent whole-body dose to an adult from public water supply from radium (U.S. Environmental Protection Agency 1976a)
- 4.3×10^{-4} Sv/yr, whole-body exposure for home construction on phosphate lands (U.S. Environmental Protection Agency 1979)
- 9×10^{-4} Sv/yr, gamma dose from disposal of uranium wastes (U.S. Nuclear Regulatory Commission 1981f)

8.2.6. Natural Background Radiation

According to the National Council on Radiation Protection and Measurements (NCRP), whole-body irradiation from natural sources results in dose rates to individuals in the United States varying from 0.7×10^{-3} Sv/yr to about 2×10^{-3} Sv/yr, depending on location (National Council on Radiation Protection and Measurements 1975). This includes penetrating radiation from cosmic rays, from naturally occurring radionuclides in the atmosphere and in the earth's crust and from radionuclides in the human body. The average individual whole-body dose rate from these sources in the United States is about 10^{-3} Sv/yr.

The variation of background radiation dose with location is due to several factors. Cosmic radiation intensity increases with altitude; the average whole-body dose rate from cosmic rays at an elevation of 1 mile (1.6 km) is approximately twice the sea-level dose rate of about 3×10^{-4} Sv/yr (National Council on Radiation Protection and Measurements 1975). Variations in the concentration of naturally occurring radionuclides in the earth's crust also affect the doses received.. Regional whole-body dose equivalent rates from terrestrial sources range from about 2.3×10^{-4} Sv/yr in the Atlantic and Gulf coastal plain to about 9×10^{-4} Sv/yr in the Colorado plateau. The above-average dose rates in the Colorado plateau region of the United

States are due to extensive deposits of uranium (National Council on Radiation Protection and Measurements 1975).

Internal doses vary considerably as a result of variations in radioactivity concentrations in drinking water and air due to natural terrestrial sources. In some communities, elevated levels of radionuclides in the uranium-238 decay series (primarily radium-226 and daughters) in domestic water supplies result in dose rates to organs such as bone surfaces that are several times above the average U.S. bone dose rate of about 1.2×10^{-3} Sv/yr (National Council on Radiation Protection and Measurements 1975). Exposure to respiratory tissues, and to a lesser extent to other internal organs, results from the inhalation of naturally occurring radioactive gases, primarily radon-222 produced from the decay of radium-226. This is more of a problem inside buildings constructed on soil or fill materials containing natural uranium or thorium. The inhalation dose varies with several factors such as building ventilation rates, and lung doses that are several times the U.S. average lung dose of about 2×10^{-3} Sv/yr can occur (National Council on Radiation Protection and Measurements 1975). The organ doses given here have been adjusted upward by a factor of two to account for the increase in the quality factor for alpha radiation from the value of 10 used in the 1975 NCRP report to the current recommended value of 20 (International Commission on Radiological Protection 1977).

8.2.7. Use of Individual-Dose Criteria in Other Countries

It is evident from publications and project reports that the lifetime dose commitment to future individuals, or the average lifetime dose rate to individuals, is being calculated in other countries to assess the performance of geologic isolation systems (e.g., Wuschke et al. 1981). Limits to individual dose have been adopted by the Swiss regulatory authorities as safety protection goals (Nuclear Energy Agency 1980). In a draft report by the International Atomic Energy Agency (1982) for criteria for underground disposal, the principal numerical radiological protection criterion is specified in terms of the radiation dose to future individuals.

8.2.8. Summary and Discussion

The average lifetime dose rate to an individual of 10^{-4} Sv/yr selected as the criterion for this study is about 10 percent of the average annual effective whole-body dose equivalent from all sources of external background radiation in the United States. It is also less than half of the current average annual dose commitment from the ingestion of naturally occurring radionuclides. There is some parallel to future doses due to radionuclides released from a geologic repository, because ingestion is predicted to be the most important potential mode of exposure. The individual-dose criterion of 10^{-4} Sv/yr is well within the range of variations in the natural background dose rate.

The dose standards and criteria now applied by EPA and NRC for regulatory purposes suggest that a dose equivalent of 2.5×10^{-4} Sv/yr to the average individual in a particular exposed population group or to the maximally exposed individual is considered to be sufficient protection for public health and safety. The panel has seen several regulations or proposed regulations that use this value. We have also seen a regulation that uses 10^{-4} Sv/yr and one that uses 0.5×10^{-4} Sv/yr.

The values of 2.5×10^{-4} Sv/yr are applied to radionuclides that are, in many instances, identical with some of the more important radionuclides of concern in a geologic repository for high-level waste. Examples are the application of 2.5×10^{-4} Sv/yr to ground disposal of low-level radioactive wastes (U.S. Nuclear Regulatory Commission 1981a) and radium in drinking water (U.S. Environmental Protection Agency 1976a). Other examples are the 10^{-4} Sv/yr limit for decommissioning nuclear facilities, together with the bone-dose equivalent exposure limits of 3×10^{-5} Sv/yr for uranium and transuranic wastes.

The ICRP has concluded that an annual whole-body equivalent dose of 10^{-3} Sv/yr represents an acceptable level of risk such that no limit need be placed on the size of the exposed population (International Commission on Radiological Protection 1977). This is tenfold greater than the value selected by the panel.

Finally, several regulations or guides use 5×10^{-3} Sv/yr as an acceptable limit for exposure to individual members of the public, including the proposed NRC rule on low-level waste disposal.

These observations suggest that the panel's individual-dose criterion of 10^{-4} Sv/yr provides an adequate margin, of about two orders of magnitude, below a dose that could be reasonable cause for concern.

8.2.9. Application of the Panel's Performance Criterion

The panel wishes to make clear that the individual-dose criterion of 10^{-4} Sv/yr is <u>not</u> intended as an upper limit of radiation exposure. It is simply a goal against which a particular repository system performance can be compared. The dose value of 10^{-4} Sv/yr is sufficiently low so as to provide reasonable assurance that no member of the public will be exposed to a radiation risk greater than that experienced and permitted from natural sources in day-to-day life. Other, higher limits (International Commission on Radiological Protection 1979) should be used to evaluate the upper levels of exposure estimated from the uncertainties inherent to the parameters used for calculating system performance.

8.3. AS LOW AS REASONABLY ACHIEVABLE

The panel has made no evaluation as to whether the individual dose rate criterion of 10^{-4} Sv/yr is as low as is reasonably achievable,

referred to by the U.S. Nuclear Regulatory Commission (1982a) as ALARA. According to the International Commission on Radiological Protection (1973), determining whether radiation exposure from a proposed activity is as low as reasonably achievable requires, among other things, an assessment of the costs of protective measures weighed against the expected benefits from the activity. Present data on the design, costs, and expected performance for geologic waste disposal are not sufficient for any meaningful analysis of the costs of incremental reductions in radiation dose. As will be seen in the ensuing discussion of this chapter and in Chapter 9, there are large uncertainties in estimating the radiation dose, particularly if it is the population dose usually considered in ALARA calculations, and there are large uncertainties in the technological performance of a geologic disposal system. It is not possible to make any meaningful determination at this time of what radiation doses and dose rates would be as low as reasonably achievable.

8.4. FEATURES OF INDIVIDUAL-RISK AND POPULATION-RISK CRITERIA

A central issue is whether to select for the present study an overall performance criterion based on the radiation dose, or risk to future individuals, or on the integrated radiation dose, or risk to future populations. There are important features to each.

A population-risk criterion, if implementable, could avoid undue risk to a large number of people and over many generations. A given risk to an individual may be reasonable if it does not exceed some specified level of acceptable risk. Radiation exposures that result in risks not far below the specified acceptable limits for individuals may be unacceptable if large numbers and many generations of people are similarly exposed. Some geologic repositories for radioactive waste are predicted to release small quantities of radionuclides over long periods of time, and there are mechanisms by which this released radioactivity can reach even distant populations. This argues for serious consideration of a population-risk-based criterion for geologic repositories.

If such a population-risk limit could be meaningfully implemented, it would be reasonable to accept an allowable dose to a few individuals that is greater than the average lifetime individual dose to the population. However, a population-risk criterion alone, as is proposed by EPA for geologic repositories, can allow intolerable risk to a few individuals, as is shown in the ensuing analyses in this chapter and in Chapter 9.

The practical differences between an individual-risk criterion and a population-risk criterion can be illustrated for a geologic repository for which the expected risk results from the long-term groundwater transport of a small portion of the radioactive inventory to the environment. Future humans can use that contaminated water for drinking and for growing food. A future maximally exposed individual is conservatively assumed to be one who obtains his lifetime intake of potable water from this contaminated source and his lifetime intake of food from edible species grown in or by this contaminated water. This

lifetime radiation dose can be estimated from estimated concentrations of released radionuclides in water and from food chain calculations. By assuming that risk is proportional to accumulated dose, the calculated individual dose is translated to an estimated risk to that individual. This assumption is the linear hypothesis that has been adopted by the Committee on the Biological Effects of Ionizing Radiation (BEIR)(National Research Council 1980) for estimating stochastic risks that might be associated with the lifetime exposure of a population of individuals to radiation at a dose equivalent rate of about 10^{-2} Sv/yr or greater, provided the exposures are well below the range of acute exposures where the effects become nonstochastic. Adoption of this linear hypothesis for all levels of lifetime exposure that could result from a geologic repository is a fundamental assumption in EPA's proposed standard (U.S. Environmental Protection Agency 1982).

There are many uncertainties in such a calculation of maximum individual dose, including uncertainties in the prediction of waste dissolution and hydrogeologic transport. The dose calculation assumes that future individuals will have dietary habits similar to those at present. It assumes that they will drink about the same amount of water, will eat the same amount and kind of food, and will respond in the same way to ingested radionuclides. It assumes that agricultural practices and uptake of radionuclides in food chains will be the same as at present. The result of a criterion based on individual dose is to limit the maximum concentration of individual radionuclides in water contaminated by future radioactive releases from a repository. This maximum concentration can be controlled by choosing radioactive waste material with suitably small rates of dissolution, by selecting repository sites with suitably long travel times for contaminated groundwater to reach the environment, and by selecting repository sites with natural processes that dilute the concentration of released radionuclides.

Once a population-risk criterion has been specified, to determine compliance, such as the maximum number of health effects from the entire exposed population over some specified number of generations, it will be necessary, in principle, to integrate the individual radiation exposures and risks therefrom over extremely large numbers of people and over many generations. Some of these individuals may be the maximally exposed individuals described above who live in the vicinity of the repository site; others will live elsewhere but will consume some of the food products grown from the contaminated groundwater. A large number of these individuals will receive extremely small radiation doses.

To estimate each individual risk, it is necessary to know the relation between individual dose and risk for individual doses ranging from the minuscule to the maximum described above. No proved or recommended relation exists over this dose range (cf. Section 8.5.7). However, if one assumes, as does EPA, that the linear hypothesis is valid for all values of the incremental collective (i.e., accumulated) dose to the many different exposed individuals, then the estimate of population risk translates to an estimate of the total radiation dose summed over all of the exposed people for many generations.

A meaningful estimate of the total intergenerational population dose requires detailed estimates of the time-dependent number of individuals who may be living in the vicinity of the repository site and the extent to which their water and food are derived from the contaminated water, and it requires estimates of the extent to which this contaminated water is used to grow food for shipment to other parts of the world. It requires estimates of the time-dependent numbers, geographical distribution, and eating habits of the future populations who may eat food grown by the contaminated water. Clearly, the direct estimate of population dose and risk involves far more uncertainties than does the estimate of the dose and risk to the maximally exposed individual.

If the population-risk criterion is translated by the regulator into a limit on the amount of radionuclides released from a repository, as has been attempted by EPA (1982), then the derived release limit is subject to all of the uncertainties described above for determining compliance with a population limit, and the uncertainties are compounded by the need to assume how radioactivity released from presently unspecified repository sites will result in radiation doses to future humans. These problems inherent in EPA's derivation of release limits to achieve a specified population-dose criterion are discussed in Section 8.5.

The implementation of a population-risk criterion, or of a release-limit criterion derived therefrom, could result in a different dependency of risk on the concentration of radionuclides in the contaminated water. For example, if a low radionuclide concentration results because of dilution in water at the repository site, the dose to the maximally exposed individual is reduced, but there may be a larger number of future individuals who use the contaminated water. If it can be assumed that, within a given generation, the number of people using the contaminated water or ingesting food grown from the contaminated water is proportional to the volume or flow rate of the contaminated water, then a population-dose limit for that generation would result in a limit on the amount of radionuclides released to the environment during a human lifetime. It would not limit the concentration of these radionuclides in contaminated water.

If an intergenerational population-dose or risk criterion is to be applied, and if parameters assumed above to be constant over a human lifetime are now assumed to be constant over the many tens and hundreds and thousands of years found to be important in estimating individual doses, then again a population-dose limit would not limit the concentration of radionuclides in contaminated water but would limit the total amount released over the long time period of many generations.

Application of a population-dose criterion can result in views as to the adequacy of a repository site that can be quite different from those resulting from an individual-dose criterion. Large differences in estimated doses to maximally exposed individuals for different repository sites are shown in Sections 9.7 and 9.10, where the different individual doses result mainly from differences in water flow rates. The contrast between different sites is less apparent if a population-dose criterion is applied, without considering doses to individuals.

In principle, dose criteria that specify both the dose to the maximally exposed individual and the intergenerational dose to future populations should be considered. However, the latter should be adopted as a numerical criterion only if it can be reasonably implemented without requiring the accumulation of unrealistic assumptions that would undermine its validity. If this proves to be impossible, a semiquantitative or qualitative criterion may still serve the purpose of limiting the integrated risk over many generations to a tolerable level. This would help emphasize that compliance with an individual-risk criterion should be achieved to a maximum reasonable extent by containment and decay within the geologic disposal system.

In past regulatory practice (e.g., 10 CFR 20, Sect. 106(e)), it has frequently been assumed that the choice of a suitably conservative individual-dose criterion, such as that adopted in this study, would result in acceptably low population doses. The same principle may apply to geologic repositories, although this assumption would require some detailed study.

8.5. THE EPA PROPOSED STANDARD

8.5.1. Introduction

The proposed EPA standard has undergone an evolutionary process. Draft number 12 was provided to the panel in September 1980. Meetings and extensive written communications have been held with members of the EPA staff to discuss their technical work that forms the basis of the proposed standard. Several drafts have been provided to the panel during the course of the study, and the proposed standard was released for public comment in late 1982 (U.S. Environmental Protection Agency 1982). Our technical evaluation is based on these drafts, on discussions and communications with the EPA staff, and on EPA draft technical reports (C. B. Smith et al. 1981, J. M. Smith et al. 1981) provided to the panel for the purpose of this review. Additional draft reports by the EPA staff are understood to be pertinent to this review, but they were not provided during the panel study. These include an individual-dose assessment report, an environmental impact statement, and other identified reports (D. J. Egan, Environmental Protection Agency, personal communication to T. H. Pigford, 1981). The review summarized herein is based in part on a more detailed technical analysis by Pigford and Mann (1982) carried out to support the panel's study.

A fundamental premise of the proposed EPA standard is that there should be no more than 1,000 fatalities ("health effects") in the next 10,000 years resulting from radioactive releases from a full-scale geologic repository, i.e., a repository containing the radioactive inventory from 100,000 Mg of uranium fuel from light-water reactors. EPA proposes to achieve that objective by prescribing quantitative limits on the amounts of radioactivity from various radionuclides that could be released to the "accessible environment" over a period of 10,000 years.

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penetrate the backfill material that might surround the waste. One quoted study calculates dissolution rates averaged over several thousand years, including the time during which no radionuclides have penetrated the backfill. If NRC intends that its release rate criterion be anything other than a performance standard that must be met at every instant, it should so state. To assess the ability of a backfill material to help attain NRC's required release rate, the rate of release of long-lived radionuclides from the waste package during the eventual steady state diffusion through the backfill should be considered, i.e., during the time when backfill sorption has little effect on the release rate.

We do not consider the NRC staff's review of the waste-form technology and its expected and possible performance in a repository to be adequate to support their proposed release rate criterion. The NRC evaluation is not based on a predictive technique with a clear or valid technical basis. To verify that waste packages comply with such performance criteria in the long-term future, a proven and reliable predictive technique must be available. Such verification has not been addressed in the proposed rule, and the predictive techniques used by NRC do not seem capable of verification.

8.6.10. Summary

The NRC's numerical criteria for 1,000-year containment and an across-the-board fractional release rate of 10^{-5}/yr from the waste package after 1,000 years are of questionable importance to long-term safety and are proposed without a technically valid basis and with invalid assumptions of existing technology. If such numerical criteria were adopted, compliance could probably not be verified. It would be more appropriate for NRC to state the considerations that may help guide DOE in its development and proof of the waste package as one of the possible barriers that may aid in meeting a reasonable overall safety criterion. One of the important considerations is for DOE to continue work on developing a means of predicting the long-term performance of waste packages.

NRC has not yet addressed the question of how either its numerical criteria or the EPA release limits can "support a finding of no unreasonable risk to the health and safety of the public," the objective stated in NRC's first release of the proposed rule.

Because of the foregoing, we do not adopt the NRC technical criteria as indicators of satisfactory performance of the waste-isolation technology. Instead, we adopt the overall performance criterion in terms of average annual radiation dose to an individual, and in Chapter 9 we evaluate the expected and possible performance of the waste-isolation system in terms of meeting this overall criterion.

8.7. IS GEOLOGIC ISOLATION INCOMPATIBLE WITH AN INDIVIDUAL-DOSE CRITERION?

It has been suggested that adopting an individual-dose criterion for geologic waste isolation can lead to approaches for protecting future individuals from released radioactivity that may be incompatible with achieving waste isolation (M. J. Bell, Nuclear Regulatory Commission, personal communication to T. H. Pigford, 1982). As an example, the magnitude of individual dose from ingestion depends on the concentration of radionuclides in water, so one way of reducing individual doses is to dilute released radionuclides with large quantities of water. By contrast, a stated goal of geologic isolation is to prevent radionuclides from reaching the environment, so that techniques of dilution are unnecessary. The problem occurs because there is no complete isolation of all radionuclides in geologic systems. Geologic repositories in natural salt come the closest because there is no flowing groundwater in salt to carry radionuclides to the environment. Isolation in natural salt would seem to be complete but for the possible intrusion into the salt repository by people or by a major natural diversion of flowing water, however improbable it may be. Therefore, we must deal with the reality of incomplete isolation, and the public must be protected from those radionuclides that eventually do reach the environment.

We have already pointed out in Section 8.4 that although limiting the total releases of radioactivity to the environment, as in the EPA and NRC proposals, might seem the most direct and practical approach toward achieving isolation, the proposed standard with its 10,000-year release limits does little toward limiting the releases that will later occur. Further, our calculations summarized in Chapter 9 show that even though the EPA 10,000-year release limits are achieved by the NRC numerical criteria, the radionuclides released to the groundwater that EPA and NRC intend to protect are likely to reach concentrations far greater than would be allowed in potable water. We show that radionuclides released to surface waters can be easily diluted by rapidly flowing rivers to concentrations far below the levels that would result even in the low individual-dose criterion of 10^{-4} Sv/yr adopted for this study. We also show that for surface water flowing at a relatively low rate, the released radionuclides may not be diluted enough to result in sufficiently low radiation doses to individuals. Although these individual doses are ignored in the proposals by EPA and NRC, protecting the health and safety of future individuals is not to be ignored.

What this means is that geologic isolation is a worthwhile objective, but it should not become a slogan that obscures the fact that even though most of the radionuclides will be successfully isolated long enough to disappear by decay, some will be released. Protection of individuals and populations from those released radionuclides is a realistic, legitimate, and nontrivial issue. If hazards to individuals

from released radionuclides can be reduced by taking advantage of dilution, as by rapidly flowing rivers, then repository sites that need and have that capability for dilution should be given full credit.

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Section 4 - Individual and Groundwater Protection

ENVIRONMENTAL PROTECTION AGENCY

40 CFR Parts 141, 142

[WH-FRL 3956-4]

RIN 2040-AA94

National Primary Drinking Water Regulations; Radionuclides

AGENCY: Environmental Protection Agency.

ACTION: Notice of proposed rulemaking.

SUMMARY: In this action under the Safe Drinking Water Act (as amended in 1986), the Environmental Protection Agency (EPA) is proposing Maximum Contaminant Level Goals (MCLGs) and National Primary Drinking Water Regulations for the following radionuclides: radon-222, radium-226, radium-228, uranium, alpha emitters, and beta particle and photon emitters. These radionuclides are classified as group A human carcinogens according to EPA's classification scheme; also, uranium is toxic to the kidneys. This notice proposes MCLGs, Maximum Contaminant Levels (MCLs), monitoring, reporting, and public notification requirements for these radionuclides.

DATES: Written comments should be submitted by October 16, 1991. A public hearing will be held on September 6, 1991 in Washington, DC beginning at 9 a.m. A second public meeting will be held on September 12, 1991 in Chicago, Illinois at 9 a.m. Washington hearing speakers should register by August 23. Chicago hearing speakers should register by August 30.

ADDRESSES: Send written comments to Comments Clerk—Radionuclides, Drinking Water Standards Division, Office of Ground Water and Drinking Water (WH-550D), Environmental Protection Agency, 401 M Street, SW., Washington, DC 20460. A copy of all public comments and supporting documents for this proposed regulation will be available for review at EPA, Ground Water and Drinking Water Docket, 401 M Street, SW., Washington, DC 20460. For access to the docket materials, call 202-382-3027 between 9 a.m. and 3:30 p.m. Commenters are requested to submit one original and three copies of their written comments. Commenters who wish to receive acknowledgement of receipt of their comments should include a self addressed stamped envelope. All comments must be post marked or delivered by hand by October 16, 1991. No facsimiles (faxes) will be accepted, as EPA is not equipped to receive the large volume of comments expected to arrive near the close of the comment period, and cannot assure that faxes will be delivered to the docket. Major supporting documents cited in the reference section of the proposed rule will be available for inspection at the Drinking Water Supply Branches in EPA's Regional Offices listed below:

I. JFK Federal Bldg., (One Congress Street, 11th floor), Boston, MA 02203, Phone: (617) 565-3610, Jerome Healey
II. 26 Federal Plaza, Room 624, New York, NY 10278, Phone: (212) 264-1800, Walter Andrews
III. 841 Chestnut Street, Philadelphia, PA 19107, Phone: (215) 597-9873, Dale Long
IV. 345 Courtland Street, Atlanta, GA 30365, Phone: (404) 347-3633, Wayne Aeronson
V. 230 S. Dearborn Street, Chicago, IL 60604, Phone: (312) 353-2650, Ed Watters
VI. 1445 Ross Avenue, Dallas, TX 75202, Phone: (214) 655-7155, Thomas Love
VII. 726 Minnesota Avenue, Kansas City, KS 66101, Phone: (913) 236-2815, Ralph Langemeir
VIII. One Denver Place, 9999 18th Street, Suite 1300 Denver, CO 80202-2413, Phone: (303) 293-1424, Patrick Crotty
IX. 75 Hawthorne Street, San Francisco, CA 94105, Phone: (415) 974-8073, Bruce Macler
X. 1200 Sixth Avenue, Seattle, WA 98101, Phone: (206) 442-1225, Jan Hastings

Public hearings will be held in the following locations:

Washington DC—Crystal City Marriott Hotel, 1111 Jefferson Davis Highway, Arlington, VA

Chicago, Illinois—J.C. Kluczynski Federal Building, 230 Dearborn Street, 16th Floor, Chicago, IL

Members of the public who plan to make a statement at either public hearing should contact Danesha Reid to register, EPA (WH-550D), 401 M Street, SW., Washington, DC 20460, telephone (202) 382-7575. Unregistered speakers will be heard after all registered speakers have made their statements.

FOR FURTHER INFORMATION CONTACT: The Safe Drinking Water Hotline, telephone (800) 426-4791, or Gregory Helms, Drinking Water Standards Division, Office of Ground Water and Drinking Water (WH-550D), Environmental Protection Agency, 401 M Street, SW., Washington, DC 20460, telephone (202) 382-7575.

Abbreviations Used in This Notice

BAT: Best Available Technology
BEIR: Committee on the Biological Effects of Ionizing Radiation
CWS: Community Water System
EMSL: EPA Environmental Monitoring and Support Laboratory (Cincinnati or Las Vegas)
ede: effective dose equivalent
GAC: Granular Activated Carbon
ICRP: International Commission on Radiation Protection
MCL: Maximum Contaminant Level
MCLG: Maximum Contaminant Level Goal
MDL: Method Detection Limit
Mr/hr: milliroentgen per hour
mgd: Million Gallons/Day
mrem/yr: millirem/year
NIPDWR: National Interim Primary Drinking Water Regulation
NPDWR: National Primary Drinking Water Regulation
NTNC: Non-transient, non-community water system
pCi/l: picocurie/liter
POE: Point-of-Entry Technologies
POU: Point-of-Use Technologies
PQL: Practical Quantitation Level
PTA: Packed Tower Aeration
PWS: Public Water System
Ra-226: Radium-226
Ra-228: Radium-228
RIA: Regulatory Impact Analysis
Rn-222: Radon-222, or radon
SDWA: Safe Drinking Water Act, or the "Act", as amended in 1986
SMR: Standard Mortality Ratio
WLM: Working Level Month

Table of Contents

I. Summary of Today's NPRM

Applicability

The regulations proposed in this notice would apply to all community and all non-transient, non-community public water systems. The proposed regulations would not apply to private water supplies (i.e., systems serving fewer than 25 persons).

Proposed MCLGs and MCLs

	MCLG	MCL
1. Radium-226	zero	20 pCi/l.
2. Radium-228	zero	20 pCi/l.
3. Radon-222	zero	300 pCi/l.
4. Uranium	zero	20 μg/l (30 pCi/l).
5. Beta and photon emitters (excluding Ra-226).	zero	4 mrem ede/yr.
6. Adjusted gross alpha emitters (excluding Ra-226, U, and Rn-222).	zero	15 pCi/l.

Notes: EPA recognizes that most radionuclides emit more than one kind of radiation as they decay. The lists of compounds labeled "alpha" or "beta" emitters identify the predominant mode of decay.

Notes: In this document the unit mrem ede/yr refers to the dose committed over a period of 50 years to reference man (ICRP 1975) from an annual intake at the rate of 2 liters of drinking water per day.

Proposed BATs Under Section 1412 of the SDWA

Radium 226/228: Ion exchange, lime softening, reverse osmosis
Radon: Aeration
Uranium: Coagulation/filtration, ion exchange, lime softening, reverse osmosis
Beta and photon emitters: Ion exchange, reverse osmosis
Alpha emitters: Reverse osmosis

Proposed BAT Under Section 1415 of the SDWA

The same as BAT under Section 1412. Coagulation and filtration and lime softening are not BAT for small systems (those with ≤500 connections) for the purpose of granting variances because they are not technologically feasible for small systems.

Proposed Compliance Monitoring

(a) The proposed initial monitoring requirements for radon are:

(1) For ground water systems and mixed ground and surface water systems, four consecutive quarterly samples for one year, and then annual samples for the remainder of the first three year compliance period. States could grant monitoring waivers to systems that demonstrate compliance with the MCL reliably and consistently in the initial compliance period, allowing systems to collect only one sample per three year compliance period for the remainder of the nine year compliance cycle. Systems relying solely on surface water are not required to monitor for radon, because radon is a highly volatile gas and is not expected to be found in surface water. Laboratories would be expected to accurately measure radon down to levels of 300 pCi/l at the time of sampling.

(2) Systems that violate the MCL would be required to monitor quarterly until the average of four consecutive quarterly samples is below the MCL.

(b) The proposed monitoring requirements for gross alpha, radium and uranium are:

(1) Three annual gross alpha screens, to be initiated in the compliance period starting January 1996; if gross alpha is less than the MCLs for radium-226, uranium, and adjusted gross alpha, screening would be reduced to monitoring once per three year compliance period. Laboratories would be expected to measure radium 226 and uranium down to 3 pCi/l and gross alpha down to 15 pCi/l.

(2) If gross alpha exceeds the radium-226, uranium, or adjusted gross alpha MCLs, specific analysis for uranium and/or radium-226 must be conducted. If the contaminant-specific analyses show that the radium-226 or uranium MCL was exceeded, quarterly monitoring for that contaminant is required. If neither MCL is exceeded, monitoring for radium-226 and uranium (or gross alpha screen in lieu of radium or uranium) may be reduced to one sample every 3-year compliance period after 3 annual samples. Sampling may be reduced to one sample every 9-year compliance cycle if the state finds, through a monitoring waiver, that the system meets the MCL reliably and consistently.

(3) Systems that violate the MCL would be required to monitor quarterly until four consecutive quarterly samples is below the MCL.

(c) The proposed monitoring requirements for radium-228 are as follows: Three annual radium-228 analyses would be required; if the radium-228 MCL is exceeded, quarterly monitoring would be required. If the system is consistently below the MCL, then the annual period may be reduced to one sample per three year compliance period. Monitoring may be further reduced to once every 9-year compliance cycle by the issuance of a monitoring waiver if the state finds that the system meets the MCL reliably and consistently. A gross beta test may be used as a screen for radium 228. Systems that violate the MCL would be required to monitor quarterly until four consecutive quarterly samples is below the MCL.

(d) Gross beta monitoring. Only supplies deemed vulnerable to contamination would be required to monitor for beta and photon emitters. Vulnerable systems would be required to measure gross beta quarterly and tritium and strontium annually. The presumptive screen for compliance with the MCL would be 50 pCi/l. Because

Greek, prefix & abbreviation	Value	Shorthand-exponential notation	Description
milli—m	1/1,000	10^{-3}	One part per thousand.
micro—Greek m	1/1,000,000	10^{-6}	One part per million.
nano—n	1/1,000,000,000	10^{-9}	One part per billion.
pico—p	1/1,000,000,000,000	10^{-12}	
femto—f	1/1,000,000,000,000,000	10^{-15}	
atto—a	1/1,000,000,000,000,000,000	10^{-18}	

Thus 1 picocurie is a millionth millionth of a curie and is abbreviated 1 pCi. Also 1 millirad (1 mrad) is one thousandth of a rad.

Because of the particle mass and charge, 1 rad deposited in tissue by alpha particles creates a more concentrated biological damage than 1 rad of gamma rays. To compensate for this difference in damage and subsequent effect, a new unit was created—the rem. This is called the dose equivalent. The absorbed dose is measured in rads and the dose equivalent is measured in rems.

The rad and rem are related by a quality factor as follows:

Number of rems = Q times the number of rads

Where Q is the quality factor which has been assigned the following value:

Q = 1 for beta particles and all electromagnetic radiations (gamma rays and x-rays)

Q = 10 for neutrons from spontaneous fission and for protons

Q = 20 for alpha particles and fission fragments

The quality factor is meant to approximately account for the relative harm caused by various types of radiation. The International System (SI) unit corresponding to the rem is the Sievert (Sv). One Sievert equals 100 rem.

Appendix B—Beta Particle and Photon Emitters

Nuclide	Ch (pCi/liter)
H-3	6.09E+04
BE-7	4.35E+04
N-13	1.52E+05
C-11	9.92E+04
C-14	3.20E+03
C-15	6.69E+06
O-15	4.95E+06
F-18	3.95E+04
NA-22	4.66E+02
NA-24	3.35E+03
SI-31	1.02E+04
P-32	6.41E+02
P-33	1.87E+03
S-35	1.29E+04
CL-36	1.85E+03
CL-38	2.12E+04
K-42	3.90E+03
CA-45	1.73E+03
CA-47	8.46E+02
SC-46	6.63E+02
SC-47	2.44E+03
SC-48	7.66E+02
V-48	6.44E+02
CR-51	3.80E+04
MN-52	7.30E+02
MN-54	2.01E+03
MN-56	5.64E+03
FE-55	9.25E+03

Appendix B—Beta Particle and Photon Emitters—Continued

Nuclide	Ch (pCi/liter)
FE-59	8.44E+02
CO-57	4.87E+03
CO-58	1.59E+03
CO-58M	6.49E+04
CO-60	2.18E+02
NI-59	2.70E+04
NI-63	8.91E+03
NI-65	8.81E+03
CU-64	1.19E+04
ZN-65	3.96E+02
ZN-69	6.31E+04
ZN-69M	4.22E+03
GA-67	7.02E+03
GA-72	1.19E+03
GE-71	4.36E+05
AS-73	7.85E+03
AS-74	1.41E+03
AS-76	1.06E+03
AS-77	4.53E+03
SE-75	5.74E+02
BR-82	3.15E+03
RB-82	4.36E+05
RB-86	4.85E+02
RB-87	6.01E+02
RB-88	2.91E+04
RB-89	5.27E+04
SR-82	2.41E+02
SR-85	2.83E+03
SR-85M	2.37E+05
SR-89	5.99E+02
SR-90	4.20E+01
SR-91	2.16E+03
SR-92	3.10E+03
Y-90	5.10E+02
Y-91	5.76E+02
Y-91M	1.32E+05
Y-92	2.57E+03
Y-93	1.20E+03
ZR-93	5.09E+03
ZR-95	1.46E+03
ZR-97	6.50E+02
NB-93M	1.05E+04
NB-94	7.07E+02
NB-95	2.15E+03
NB-95M	2.39E+03
NB-97	2.35E+04
NB-97M	1.37E+06
MO-99	1.83E+03
TC-95	6.97E+04
TC-95M	3.12E+03
TC-96	2.05E+03
TC-96M	1.76E+05
TC-97	3.25E+04
TC-97M	4.45E+03
TC-99	3.79E+03
TC-99M	8.96E+04
RU-97	7.96E+03
RU-103	1.81E+03
RU-105	4.99E+03
RU-106	2.03E+02
RH-103M	4.71E+05
RH-105	3.72E+03
RH-105M	5.51E+06

Appendix B—Beta Particle and Photon Emitters—Continued

Nuclide	Ch (pCi/liter)
RH-106	1.24E+06
PD-100	1.30E+03
PD-101	1.34E+04
PD-103	6.94E+03
PD-107	3.66E+04
PD-109	2.12E+03
AG-105	2.70E+03
AG-106	6.26E+05
AG-106M	7.23E+02
AG-109M	1.57E+07
AG-110	1.84E+06
AG-110M	5.12E+02
AG-111	1.06E+03
CD-109	2.27E+02
CD-115	8.58E+02
CD-115M	3.39E+02
IN-113M	5.24E+04
IN-114	9.76E+05
IN-114M	3.23E+02
IN-115	3.51E+01
IN-115M	1.64E+04
SN-113	1.74E+03
SN-121	6.06E+03
SN-121M	2.26E+03
SN-125	4.46E+02
SN-126	2.93E+02
SB-122	8.10E+02
SB-124	5.63E+02
SB-125	1.94E+03
SB-126	5.44E+02
SB-126M	5.85E+04
SB-127	6.18E+02
SB-129	3.09E+03
TE-125M	1.49E+03
TE-127	7.92E+03
TE-127M	6.53E+02
TE-129	2.72E+04
TE-129M	5.24E+02
TE-131	2.68E+04
TE-131M	9.71E+02
TE-132	5.80E+02
I-122	2.11E+05
I-123	1.07E+04
I-125	1.51E+02
I-126	8.10E+01
I-129	2.10E+01
I-130	1.19E+03
I-131	1.06E+02
I-132	8.19E+03
I-133	5.49E+02
I-134	2.14E+04
I-135	2.34E+03
CS-131	2.28E+04
CS-134	8.13E+01
CS-134M	1.01E+05
CS-135	7.94E+02
CS-136	5.18E+02
CS-137	1.19E+02
CS-138	2.56E+04
BA-131	2.95E+03
BA-133	1.52E+03
BA-133M	2.62E+03
BA-137M	2.15E+06

APPENDIX B—BETA PARTICLE AND PHOTON EMITTERS—Continued

Nuclide	Ch (pCi/liter)
BA-139	1.38E+04
BA-140	5.82E+02
LA-140	8.52E+02
CE-141	1.89E+03
CE-143	1.21E+03
CE-144	2.61E+02
PR-142	1.04E+03
PR-143	1.17E+03
PR-144	4.70E+04
PR-144M	1.12E+06
ND-147	1.25E+03
ND-149	1.17E+04
PM-147	5.24E+03
PM-148	5.05E+02
PM-148M	5.75E+02
PM-149	1.38E+03
SM-151	1.41E+04
SM-153	1.83E+03
EU-152	8.41E+02
EU-154	5.73E+02
EU-155	3.59E+03
EU-156	8.00E+02
GD-153	4.68E+03
GD-159	2.76E+03
TB-158	1.25E+03
TB-160	8.15E+02
DY-165	1.51E+04
DY-166	8.30E+02
HO-166	9.81E+02
ER-169	3.64E+03
ER-171	3.80E+03
TM-170	1.03E+03
TM-171	1.27E+04
YB-169	1.83E+03
YB-175	3.11E+03
LU-177	2.55E+03
HF-181	1.17E+03
TA-182	8.42E+02
W-181	1.90E+04
W-185	3.44E+03
W-187	2.66E+03
RE-183	5.40E+03
RE-186	1.88E+03
RE-187	5.82E+05
RE-188	1.79E+03
OS-185	2.46E+03
OS-191	2.38E+03
OS-191M	1.43E+04
OS-193	1.69E+03
IR-190	1.01E+03
IR-192	9.57E+02
IR-194	1.04E+03
PT-191	3.81E+03
PT-193	4.81E+04
PT-193M	3.02E+03
PT-197	3.40E+03
PT-197M	1.75E+04
AU-196	3.66E+03
AU-198	1.31E+03
HG-197	5.76E+03
HG-203	2.39E+03
HG-202	3.84E+03
TL-204	1.68E+03
TL-207	4.00E+06
TL-208	2.83E+06
TL-209	3.58E+05
PB-203	5.06E+03
PB-209	2.53E+04
PB-210	1.01E+00
PB-211	1.28E+04
PB-212	1.23E+02
PB-214	1.18E+04
BI-206	6.56E+02
BI-207	1.01E+03
BI-212	5.20E+03
BI-213	1.50E+04
BI-214	1.89E+04
FR-223	3.41E+03

APPENDIX B—BETA PARTICLE AND PHOTON EMITTERS—Continued

Nuclide	Ch (pCi/liter)
RA-225	9.14E+00
RA-228	7.85E+00
AC-227	1.27E+00
AC-228	3.27E+03
TH-231	4.07E+03
TH-234	4.01E+02
PA-233	1.51E+03
PA-234	2.56E+03
PA-234M	9.30E+05
U-237	1.78E+03
U-240	1.54E+03
NP-236	5.96E+03
NP-238	1.39E+03
NP-239	1.68E+03
NP-240	2.31E+04
NP-240M	1.74E+05
PU-241	6.26E+01
PU-243	1.64E+04
AM-242M	1.27E+00

Ch=Concentration in water for 4 mrem ede/y, assuming 2 liters daily intake.

APPENDIX C—ALPHA EMITTERS

NUCLIDE	Cm (pCi/liter)	Ci (pCi/liter)
SM-147	1.06E+02	1.04E+02
BI-210	1.94E+03	1.01E+03
BI-211	2.05E+05	1.56E+05
PO-210	1.40E+01	7.45E+00
PO-212	1.15E+14	8.78E+13
PO-213	8.03E+12	8.06E+12
PO-214	2.43E+11	1.86E+11
PO-215	9.17E+09	6.84E+09
PO-216	7.38E+07	5.30E+07
PO-218	9.50E+04	6.91E+04
AT-217	5.74E+06	4.27E+06
FR-221	4.50E+04	3.26E+04
RA-223	3.21E+01	2.41E+01
RA-224	5.46E+01	4.06E+01
RA-226	2.07E+01	1.57E+01
AC-225	1.85E+02	1.13E+02
TH-227	8.62E+02	4.03E+02
TH-228	1.53E+02	1.25E+02
TH-229	5.15E+01	4.93E+01
TH-230	8.27E+01	7.92E+01
TH-232	9.16E+01	8.80E+01
PA-231	1.02E+01	1.02E+01
U-232	1.02E+01	5.72E+00
U-233	2.56E+01	1.38E+01
U-234	2.59E+01	1.39E+01
U-235	2.85E+01	1.45E+01
U-236	2.74E+01	1.47E+01
U-238	2.82E+01	1.46E+01
NP-237	7.19E+00	7.06E+00
PU-236	3.33E+01	3.23E+01
PU-238	7.15E+00	7.02E+00
PU-239	6.49E+01	6.21E+01
PU-240	6.49E+01	6.22E+01
PU-242	6.83E+01	6.54E+01
PU-244	7.02E+00	6.87E+00
AM-241	6.45E+00	6.34E+00
AM-242	8.66E+03	5.34E+03
AM-243	6.49E+00	6.37E+00
CM-242	1.45E+02	1.33E+02
CM-243	8.47E+00	8.30E+00
CM-244	1.00E+01	9.84E+00
CM-245	8.35E+00	8.23E+00
CM-246	8.38E+00	8.27E+00
CM-247	6.93E+00	6.79E+00
CM-248	1.71E+00	1.67E+00
CF-252	1.70E+01	1.62E+01

Cm=Concentration in water for lifetime mortality risk=1×10^{-4}

Ci=Concentration in water for lifetime incidence risk=1×10^{-4}

Both assume 2 liters daily intake of water.

List of Subjects in 40 CFR Parts 141 and 142

Chemicals, Reporting and record keeping requirements, Water supply, Administrative practice and procedure.

Dated: June 17, 1991.

William K. Reilly,

Administrator, Environmental Protection Agency.

For the reasons set forth in the preamble, title 40 of the Code of Federal Regulations is proposed to be amended as follows:

PART 141—NATIONAL PRIMARY DRINKING WATER REGULATIONS

1. The authority citation for part 141 continues to read as follows:

Authority: 42 U.S.C. 300f, 300g-1, 300g-2, 300g-3, 300g-4, 300g-5, 300g-6, 300j-4 and 300j-9.

2. Section 141.2 is amended by adding, in alphabetical order, a definition for "adjusted gross alpha" as follows:

§ 141.2 Definitions

• • • • •

Adjusted gross alpha: Adjusted gross alpha is defined as the result of a gross alpha measurement, less radium-226 and less uranium. Radon is not included in adjusted gross alpha.

• • • • •

3. Section 141.15 is amended by revising the introductory text to read as follows:

§ 141.15 Maximum contaminant levels for radium-226, radium-228, and gross alpha particle radioactivity in community water systems.

The following are the maximum contaminant levels for radium-226, radium-228, and gross alpha particle radioactivity, which shall remain effective until [insert date 18 months after publication of the final rule in the Federal Register]:

• • • • •

4. Section 141.16 is proposed to be amended by adding introductory text to read as follows:

§ 141.16 Maximum contaminant levels for beta particle and photon radioactivity from man-made radionuclides in community water systems.

The following maximum contaminant levels shall remain effective until [insert date 18 months after publication of the final rule in the Federal Register]:

• • • • •

5. Section 141.25 is amended by revising the section to read as follows:

PROPOSED METHODOLOGY FOR RADIONUCLIDE CONTAMINANTS

aminant	Methodology	Reference (method or page number)								
		EPA [1]	EPA [2]	EPA [3]	EPA [4]	SM [5]	ASTM [6]	USGS [7]	DOE [8]	Other
Naturally occurring										
Gross alpha and beta.	Evaporation	900.0	pp. 1-3	00-01	p. 1	7110 B	D 1943-81	R-1120-76		
Gross alpha	Co-precipitation			00-02						
Radium 226	Radon emanation.	903.1	pp. 16-23	Ra-03	p. 19	7500-Ra B	D 3454-86	R-1141-76		N.Y.[9]
	Radiochemical	903.0		Ra-05						
Radium 228	Radiochemical	904.0	pp. 24-28	Ra-05	p. 19	7500-Ra D[10]		R-1142-76		N.Y.[9] N.J.[10]
Radon 222	Liquid scintillation.									913 [11] LS [12]
	Lucas cell									LC [13]
Uranium	Radiochemical	908.0				7500-U B	D 3972-82			
	Fluorometric	908.1				7500-U C	D 2907-83	R-1180-76	E-U-03	
								R-1181-76		
	Alpha spectrometry.			00-07	p. 33			R-1182-76	E-U-04	
Man-made										
Radioactive cesium.	Precipitation	901.0	pp. 4-5			7500-Cs B		R-1110-87	E-Cs-01	
Radioactive iodine.	Precipitation	902.0		I-01		7500-I B	D 2334-88			
Radioactive strontium 89, 90.	Precipitation	905.0	pp. 29-33		p. 65	7500-Sr B		R-1160-76		
	Radiochemical		pp. 108-114	Sr-04					E-Sr-01	
Tritium	Liquid scintillation.	906.0	pp. 34-40	H-02	p. 67	7500-3H B	D 2476-81 (87)	R-1171-76		
Gamma and photon emitters.	Gamma ray Spectrometry.	901.1					D-3649-85		4.5.2.3	

[1] "Prescribed Procedures for Measurement of Radioactivity in Drinking Water," EPA Environmental Monitoring and Support Laboratory, Cincinnati, OH (EPA-600/4-80-032, August 1980. (EPA, 1980).
[2] "Interim Radiochemical Methodology for Drinking Water," EPA-600/4-75-008, March 1976. (EPA, 1976)
[3] System Environmental Radiation Facility, Montgomery, AL 36109, "Radiochemical Procedures Manual," EPA 520/5-84-006, August 1984. (EPA, 1984a).
Radiochemical Analytical Procedures for Analysis of Environmental Samples," EMSL-LV-0539-17, March 1979. (EPA, 1976b).
[4] "Standard Methods for the Examination of Water and Wastewater," 17th edition, American Public Health Association, American Water Works Association, Water Pollution Control Federation, 1989. (APHA, 1989).
[5] 1989 Annual Book of ASTM Standards, Vol. 11.02, American Society for Testing and Materials, 1916 Race Street, Philadelphia, Pa. 19103. (ASTM, 1989)
[6] "Methods for Determination of Radioactive Substances in Water and Fluvial Sediments," Book 5, 1989, Techniques of Water-Resources Investigations of the United States Geological Survey, Chapter A5. (USGS, 1989).
[7] Environmental Measurements Laboratory, U.S. Department of Energy, "EML PROCEDURES MANUAL, 27th edition." (DOE, 1990).
[8] "Determination of ^{226}Ra and ^{228}Ra (Ra-02), Radiological Sciences Institute Center for Research—New York State Department of Health, January 1980 (Revised June 1982). (NY State DOH, 1982).
[9] "Determination of Radium 226 in Drinking Water," State of New Jersey—Department of Environmental Protection—Division of Environmental Quality—Bureau of Radiation and Inorganic Analytical Services, August 1990. (NJ DEQ, 1990).
[10] Method 913—Radon in drinking water by liquid scintillation, Environmental Monitoring and Support Laboratory, Las Vegas, NV. (EPA 1991q).
[11] Appendix D, Analytical Test Procedure, "The Determinnson of Radon in Drinking Water," p. 22, Two Test Procedures for Radon in Drinking Water, Interlaboratory Collaborative Study, EPA/600/2-87/082, March 1987. (EPA, 1987e).

(2) Sample collection for radon, radium-226, radium-228, adjusted gross alpha, uranium, and beta and photon emitters under this section shall be conducted using the sample preservation, container, and maximum holding time procedures specified in the table below:

Sampling handling, preservation, holding times			
Parameter	Preservative [1]	Container [2]	Maximum holding time [3]
Gross alpha	Conc. HCl or HNO_3 to pH <2 [4]	P or G	6 months.
Gross beta	Conc. HCl or HNO_3 to pH <2 [4]	P or G	6 months.
Radium-226	Conc. HCl or HNO_3 to pH <2	P or G	6 months.
Radium-228	Conc. HCl or HNO_3 to pH <2	P or G	6 months.
Radon-222 [5]	Cool 4°C.	Glass with Teflon-lined septum	4 days
Uranium natural	Conc. HCl or HNO_3 to pH <2	P or G	6 months.
Radioactive Cesium	Conc. HCl to pH <2	P or G	6 months.
Radioactive Strontium	Conc. HCl or HNO_3 to pH <2	P or G	6 months.
Radioactive Iodine	None	P or G	6 months.
Tritium	None	G	6 months.
Photon emitters	Conc. HCl or HNO_3 to pH <2	P or G	6 months.

[1] (All except radon-22 samples). It is recommended that the preservative be added to the sample at the time of collection unless suspended solids activity is to be ... ed. However, if the sample must be shipped to a laboratory or storage area, acidification of the sample (in its original container) may be delayed for a pe... to exceed 5 days. A minimum of 16 hours must elapse between acidification and analyses
[2] P=Plastic, hard or soft; G=Glass, hard or soft

[2] Holding time is defined as the period from time of sampling to time of analysis. In all cases, samples should be analyzed as soon after collection as possible.
[3] If HCl is used to acidify samples which are to be analyzed for gross alpha or gross beta activities, the acid salts must be converted to nitrate salts before transfer of the samples to planchets.
[4] The procedure of a positive pressure collection in 60-ml glass bottles is to be followed. This procedure is described in appendix C, NIRS Sampling Instructions—Radon, p. 26, Two Test Procedures For Radon In Drinking Water, Interlaboratory Collaborative Study, EPA/600/2-87/082, March 1987.

(3) Analysis under this section shall only be conducted by laboratories that have received approval by EPA or the State. To receive approval to conduct analyses for radon, radium-226, radium-228, adjusted gross alpha, uranium, and beta and photon emitters the laboratory must:

(i) Analyze Performance Evaluation samples which include those substances provided by EPA Environmental Monitoring and Support Laboratory or equivalent samples provided by the State.

(ii) Achieve quantitative results on the analyses that are within the following acceptance limits:

Contaminant	Acceptance Limits [1]
Radium-226	±30% at ≥5 pCi/l.
Radium-228	±50% at ≥5 pCi/l.
Uranium	±30% at ≥5 pCi/l.
Radon-222 [2]	±30% at ≥300 pCi/l.
Gross alpha emitters	±50% at ≥15 pCi/l.
Gross beta emitters	±30% at ≥30 pCi/l.
Radioactive Cesium	±30% at ≥10 pCi/l.
Radioactive Iodine	±20% at ≥20 pCi/l.
Radioactive Strontium total, 89 and 90.	±30% at ≥5 pCi/l.
Tritium	±20% at ≥1200 pCi/l.

[1] Acceptance limits based on 100 minute count.
[2] Radon acceptance limits based on 4 day elapsed time from sample collection to analysis.

8. Section § 141.32 is amended by adding paragraphs (e)(77) through (82), to read as follows:

§ 141.32 Public notification.

* * * * *

(e) * * *

(77) *Radon:* The United States Environmental Protection Agency (EPA) sets drinking water standards and has determined that radon is of health concern at certain levels of exposure. Radon is a naturally occurring radioactive contaminant that occurs in ground water. It is a gas, and is released from water into household air during water use. Radon has been found in epidemiology studies to cause lung cancer in humans at high exposure levels; at lower exposure levels the risk of lung cancer is reduced. EPA has set the drinking water standard for radon in public water supplies at 300 picocuries per liter (pCi/l) to protect against lung cancer risk. Drinking water that meets the EPA standard is associated with little of this risk and is considered safe for radon.

(78) *Radium 226:* The United States Environmental Protection Agency (EPA) sets drinking water standards and has determined that radium 226 is of health concern at certain levels of exposure. Radium 226 is a naturally occurring radioactive contaminant that occurs primarily in ground water. Radium 226 has been found in epidemiology studies to cause bone cancer in humans at high exposure levels, and is believed to cause other cancers as well; at lower exposure levels the risk of cancer is reduced. EPA has set the drinking water standard for radium 226 at 20 picocuries per liter (pCi/l) to protect against cancer risk. Drinking water that meets the EPA standard is associated with little of this risk and is considered safe for radium 226.

(79) *Radium 228:* The United States Environmental Protection Agency (EPA) sets drinking water standards and has determined that radium 228 is of health concern at certain levels of exposure. Radium 228 is a naturally occurring radioactive contaminant that occurs primarily in ground water. Radium 228 has been found in epidemiology studies to cause bone cancer in humans at high exposure levels and is believed to cause other cancers as well; at lower exposure levels the risk of bone cancer is reduced. EPA has set the drinking water standard for radium 228 and 20 picocuries per liter (pCi/l) to protect against cancer risk. Drinking water that meets the EPA standard is associated with little of this risk and is considered safe for radium.

(80) *Uranium:* The United States Environmental Protection Agency (EPA) sets drinking water standards and has determined that uranium is of health concern at certain levels of exposure. Uranium is a naturally occurring radioactive contaminant that occurs in both ground and surface water. Uranium is believed to cause bone cancer and other cancers in humans at high exposure levels; at lower exposure levels the risk of cancer is reduced. EPA also believes uranium can be toxic to the kidneys. EPA has set the drinking water standard for uranium at 20 micrograms per liter (µgl) to protect against both cancer risk and risk of kidney damage. Drinking water that meets the EPA standard is associated with little of this risk and is considered safe for uranium.

(81) *Gross Alpha:* The United States Environmental Protection Agency (EPA) sets drinking water standards and has determined that alpha emitting radionuclides may be of health concern at certain levels of exposure. Alpha emitters are primarily naturally occurring radioactive contaminants, but several derive from man-made sources. They may occur in either ground or surface water. Alpha emitters are believed to cause cancer in humans at high exposure levels because they emit ionizing radiation. At lower levels, the risk of cancer is reduced. EPA has set the drinking water standard for alpha emitters at 15 picocuries per liter (pCi/l) to protect against cancer risk. Drinking water that meets the EPA standard is associated with little of this risk and is considered safe for alpha emitters.

(82) *Beta and photon emitters:* The United States Environmental Protection Agency (EPA) sets drinking water standards and has determined that beta and photon emitting radionuclides may be of health concern at certain levels of exposure. Beta and photon emitters are primarily man-made radioactive contaminants associated with the operation of nuclear power facilities, facilities using radioactive material for research or manufacturing, or facilities where these materials are disposed. Some beta emitters are naturally occurring. Beta and photon emitters are expected to occur primarily in surface water. Beta and photon emitters are believed to cause cancer in humans at high exposure levels because they emit ionizing radiation. At lower levels, the risk of cancer is reduced. EPA has set the drinking water standard for beta and photon emitters at 4 millirems effective dose equivalent per year (mrem ede/yr) to protect against cancer risk. Drinking water that meets the EPA standard is associated with little of the risk and is considered safe for beta and photon emitters.

* * * * *

7. A new section § 141.44 is added to subpart E to read as follows:

§ 141.44 Special monitoring for radionuclides.

(a) Each community and non-transient, non-community water system shall take one sample at each sampling point for lead-210 and report the results to the State. Monitoring must be completed by December 1998.

(b) Groundwater systems shall take a minimum of one sample at every entry point to the distribution system which is representative of each well after treatment (hereafter called a sampling point). Each sample must be taken at the same sampling point unless conditions

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Section 5 - Human Intrusion

HUMAN INTRUSION SCENARIOS

IN NUCLEAR WASTE REPOSITORY EVALUATIONS

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ABSTRACT

When promulgated, the Environmental Protection Agency (EPA) High Level and Transuranic Waste Standard (40 CFR 191) is expected to require consideration of human intrusion into a nuclear waste repository as part of the post-closure containment requirements. Most preliminary evaluations conclude that this will be the most difficult portion of the standard to meet. Many believe it unreasonable to require consideration of human intrusion. Others object to portions of EPA Guidance on implementation of the requirement. The Environmental Evaluation Group (EEG) believes the human intrusion requirement is reasonable, achievable, and will lead to better siting of repositories and designs of waste forms and engineered barriers.

INTRODUCTION

In September 1985 the Environmental Protection Agency (EPA) promulgated 40 CFR Part 191 - Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (the Standard). Although, there had been significant disagreements during the development stage (1978-1985) of the Standard, it was acceptable to the scientific community when promulgated.

Subpart B of the EPA Standard was subsequently vacated by a U.S. Court of Appeals in June 1987 on grounds that it was less stringent than the Safe Drinking Water Act and no explanation was provided for this discrepancy. Subpart B of the Standard was remanded to EPA for revision and repromulgation. The existence of an appropriate Standard is important to the State of New Mexico because the Waste Isolation Pilot Plant (WIPP) is located in New Mexico and has to comply with the Standard since it is intended to serve as a repository for defense transuranic waste. In August 1987 the State of New Mexico entered into an agreement (Second Modification to the 1981 Agreement for Consultation and Cooperation) with the Department of Energy (DOE) to continue to evaluate WIPP against the vacated standard. This Agreement was made because the technical requirements of the revised standards were not expected to change substantially.

During 1990, the Advisory Committee on Nuclear Waste (ACNW) of the Nuclear Regulatory Commission (NRC), the National Academy of Science Board on Radioactive Waste Management (NAS-BRWM), the Nuclear Waste Technical Review Board (NWTRB) and others have expressed doubts about the feasibility of implementing the Standard. There are several aspects of the Standard that have been criticized. One of the most significant issues is how human intrusion should be evaluated in the Standard and the assumptions that should be used. This paper discusses the human intrusion controversy, relevant aspects of the WIPP site, and the Environmental Evaluation Group's (EEG) views on the human intrusion issue.

HUMAN INTRUSION PROVISIONS IN THE STANDARD

The Standard assumes that inadvertent human intrusion into a repository from exploratory drilling for resources is possible after the active institutional control (site access control, maintenance, cleaning up releases, and monitoring) period. The Standard does not permit taking credit for active control for more than 100 years after repository closure.

Both the 1985 Standard and the 1/31/90 Working Draft 2 require the consideration of human intrusion in evaluating whether a site meets the containment requirements in subsection 191.13. It is clear from the preamble to the 1985 Standard and Appendices to the Standard and the Working Draft, that EPA intends that human intrusion be considered at all sites and that passive institutional controls (markers, records, government regulation, etc.) cannot be used to completely rule out the possibility of intrusion.

Guidance on the assumptions that may be used to evaluate human intrusion in the Standard is presented in Appendix B of the 1985 Standard and is essentially unchanged in the 1990 Working Draft. An introductory note to the Appendix states, "The supplemental information in this Appendix is not an integral part of 40 CFR 191. Therefore, the implementing agencies are not bound to follow this guidance. However, it is included because it describes the agency's assumptions regarding the implementation of Subpart B. This appendix will appear in the Code of Federal Regulations." Regardless of this disclaimer, there is a tendency on the part of those doing performance assessment to take the guidance literally and to believe that the burden-of-proof is on them if they deviate.

The Guidance states that inadvertent and intermittent intrusion by exploratory drilling for resources (other than those provided by the disposal system itself) can be the most severe intrusion scenario assumed by the implementing agencies. Also, that "implementing agencies can assume that passive institutional controls or the intruders own exploratory procedures are adequate for the intruders to soon detect, or be warned of, the incompatibility of the area with their activities."

The Guidance also suggests that "the likelihood of such inadvertent and intermittent drilling need not be taken to be greater than 30 boreholes per square kilometer of repository area per 10,000 years for geologic repositories in proximity to sedimentary rock formations, or more than 3 boreholes per square kilometer per 10,000 years for repositories in other geological formations."

Also, that "the consequences of such inadvertent drilling need not be assumed to be more severe than:

1. direct release to the land surface of all the ground water in the repository horizon that would promptly flow through the newly created borehole to the surface due to natural lithostatic pressure--of (if pumping would be required to raise water to the surface) release of 200 cubic meters of ground water pumped to the surface if that much water is readily available to be pumped; and

2. creation of a ground water flow path with the least protective practices followed by exploratory drillers (not the permeability of a borehole carefully sealed to minimize release from the repository, since the intruders may not make such an effort)."

CONTROVERSIES ABOUT HUMAN INTRUSION

A. Should Human Intrusion be included in the Standard?

Many in the scientific community believe that it is inappropriate to consider human intrusion at all in 40 CFR 191. Typical reasons given for this position are:

1. All repositories are subject to human intrusion and therefore this requirement does nothing to separate good sites from inferior ones;

2. No site could meet the standard if human intrusion must be considered;

3. It is impossible to predict the future behavior of individuals or estimate future drilling rates;

4. Society would never forget the location of a repository and would not drill in the vicinity of one. Furthermore, regulatory agencies would be in existence and forbid drilling.

There are arguments that partially refute these claims. Site locations that contain natural resources of current interest to soci are clearly less likely to be subjected to e ploratory drilling in the future than those that currently contain economic concentrations of resources. A human intrusion standard would be the strongest deterrent against choosing a resource-rich site. It remains to be seen if repository sites cannot meet a human intrusion standard even with current design concepts. Furthermore, modifications to waste and container form and repository design can be made to mitigate human intrusion consequences. Admittedly, it is difficult to predict (and impossible to prove) what future human actions will be. Historical drilling rates are a useful indication and the use of expert opinion has the potential to help predict actions or determine mitigative measures. Recent history indicates that all persons are not afraid of handling radioactive material (for example, diversions of uranium mill tailings and low level radioactive waste) and that current compliance with regulations and institutional memory are not perfect (see WIPP examples below).

B. Preliminary evaluations of human intrusion efforts

Preliminary Performance Assessment evaluations at the WIPP site and at Yucca Mountain have indicated that the consequences of hu intrusion scenarios dominate. None of

preliminary evaluations suggest that either site would have problems meeting the containment requirements in 191.13, except for human intrusion. The preliminary studies indicate that, depending on assumptions used, some human intrusion scenarios could result in the Standard being exceeded. As a result of these preliminary results a number of organizations, who appeared to be satisfied in 1985, now take exception to the human intrusion portion of the Standard.

C. Recommendations by other organizations

No organization has called for the deletion of human intrusion consideration from the Standard. However, DOE[1] has recommended that "consideration of human intrusion be separated from the CCDF" (Complementary Cumulative Distribution Function) without specifying how the human intrusion scenarios should be addressed. The ACNW[2] of the NRC has recommended "that the Standard be rewritten to separate the evaluations of anticipated performance into three parts:

1. the undisturbed repository;
2. the disturbed repository, exclusive of human intrusion; and
3. the repository as it might be affected by human intrusion.

This would clearly separate out the problem of human intrusion and permit it to be addressed directly." The ACNW did not make specific recommendations on how human intrusion should be addressed separately.

Other organizations have recommended changing some of the assumptions in the Guidance. Both Sandia National Laboratory[3] (SNL) and DOE[1] wanted the reference to maximum drilling rates (30 boreholes per square kilometer in 10,000 years) to be reworded so it was clear that this value was an upper bound and that a distribution of values less than the upper bound could be developed, justified and used. The current wording of the Guidance appears to permit the use of a distribution.

SNL[3] and DOE[1] have also taken exception to the quality of borehole sealing (following human intrusion) that should be assumed. They believe the least protective practices required by current law for exploratory drilling should constitute an upper bound and that use of a distribution of values should be permitted.

The NRC[4] staff commented that EPA reevaluate the technical base on frequency and severity of intrusion and include other than petroleum exploration. They believe the present assumptions may not be conservative.

WIPP SITE FEATURES

Conditions at WIPP maximize the frequency and effects of human intrusion. The site is located in a mineral-rich area with a history of exploration. The 30 borehole per square kilometer per 10,000 year drilling rate came from the historical drilling rate in the WIPP area. The repository horizon is located in bedded salt (the Salado Formation) at a depth of 655 meters (2150 feet). Pressurized brine reservoirs exist in the underlying Castile Formation and are believed to be present under approximately one-half of the waste storage rooms. Some of these reservoirs contain several million cubic meters of brine and have sufficient pressure to flow about 60,00 m^3 at the surface or 400,000 m^3 at the repository horizon. A brine reservoir could significantly increase the consequences of human intrusion if the brine reservoir were connected to the repository by an exploratory borehole.

Under present plans, most of the wastes coming to WIPP will not be immobilized or treated. These wastes are expected to produce large quantities of gas from organic decomposition of the waste, anoxic corrosion of metals and radiolysis. Also, there is a high void fraction inside the waste containers and in the planned salt or salt-bentonite backfill. The high void fraction coupled with potentially high gas generation and low Salado Formation gas permeability could lead to non-closure of the rooms for hundreds of years. Prior to room closure there is a potential for infiltration from Salado Formation brine, a Castile brine reservoir, or drilling fluid. Present plans are to divide the entire repository (by panel seals) into only 10 compartments.

This combination of factors leads to the possibility that a human intrusion borehole into the repository could impact several interconnected rooms where the non-fixed waste was in intimate contact with brine or where void spaces exist for the drilling fluid to circulate among the waste.

SNL is evaluating whether the present design of the repository can meet the human intrusion standard without any "enhancements," such as waste treatment and use of engineered barriers. It should not be surprising that preliminary analyses indicate the present repository design would not easily meet the Standard.

The following examples of human activities around the WIPP site indicate that the claims made by some about the degree of site control by future societies may be too optimistic:

(1) The Gnome site, located 13 kilometers (8 miles) southwest of the WIPP site, was the location of an underground

nuclear detonation in 1962. Since then the site has been cleaned up (measurable radioactivity still exists at the surface) and a monument erected. Access to the site is not restricted, the monument has degraded, and there is little awareness by area residents of this event that occurred just 28 years ago.

(2) A recent BLM study[5], which included the WIPP area, found widespread non-compliance by private industry with BLM regulations by failing to plug abandoned wells or by doing it improperly. This suggests that the SNL and DOE requests to EPA to consider that all future boreholes would meet or exceed the current standard is non-conservative.

(3) In October 1990 it was re-discovered that a gas well located just outside of the site had slant drilled under the site in 1983 and produced gas from 1983 to 1988. The lease is still active. This is interesting because DOE was aware in 1984 of the State of New Mexico's objection to deviated drilling under the site and in August, 1987 (while the well was still producing) signed an agreement with the State prohibiting slant drilling under the site. Neither the DOE nor BLM apparently remembered that the well existed between 1987 and October 1990, when a newspaper reporter received information from an anonymous source. The center of the WIPP site, where about 500 persons work, is located less than 5 kilometers (3 miles) from the surface location of this well. This incident suggests that drilling in a remote area without the knowledge of regulatory agencies hundreds of years in the future is possible.

EEG STUDIES AND POSITIONS

A. Past Evaluations

EEG has made several evaluations since 1981 that involved human intrusion. These scenarios have considered:

1. the effects of a brine reservoir interacting with the repository and the surface;[6]
2. the drilling through stacks of high-curie drums;[7]
3. the drilling into a disposal room where the waste is in a brine slurry.[7]

All of these analyses have been primar deterministic, although some of them inc porated uncertainty analysis, considered probabilities, and discussed the results in respect to the containment requirements (40 CFR 191.13). These deterministic calculations indicate that releases "might" exceed the proposed Standard. However, EEG has not considered any potential engineered enhancements (which include both waste form modification and engineered barriers) because DOE has not committed to incorporating them. Also, the potential reductions in uncertainty from on-going experiments have not been included. Our evaluations indicate that the human intrusion scenario is very significant at WIPP and the assumptions used are very important.

B. EEG Positions on Human Intrusion

EEG has expressed positions on several aspects of the human intrusion issue in the past[8,9,10]. Evaluations are currently underway which could amplify or modify some of these positions.

1. Human intrusion must be retained as an integral part of the Standard. This is especially pertinent for the WIPP site since it is located in a mineral rich area with a history exploratory drilling. Also, WIPP the presence of t requirement has forced a reevaluation of the need for engineered enhancements and for a better understanding of the repository horizon.

2. The suggested maximum drilling rates are reasonable. Those for sedimentary rock formations come from Delaware Basin (WIPP area) experience and it appears likely that exploratory drilling for hydrocarbons and other minerals will not be less in the next few centuries. Furthermore, there should be some penalty in the Standard for choosing a site in a mineral rich area.

3. The actions of drillers upon encountering waste and/or brine and in sealing boreholes should be based on current practice, not current standards or possible future technology. Recent evidence[5] indicates that oil field plugging practice does not always meet required pluggin standards. Also, that drill encountering brine often allow to flow to the surface fo extended periods.[11]

4. EEG believes the use of a distribution of values for drilling rates, drillers actions, and borehole sealing would be appropriate. Appropriate distributions to use will be difficult to arrive at. The NAS-BRWM[12] has recommended the use of expert opinion from "outside" the implementing and regulatory agencies throughout the site selection and performance assessment phase to reach agreement on assumptions to be used. EEG strongly supports this concept and believes it will be especially valuable in developing human intrusion assumptions. It is important that a consensus be sought during the performance assessment process and not "after-the-fact."

5. SNL has formed several panels of outside experts (primarily non-technical persons) to assess future events and ways that intrusion might be mitigated by monuments and barriers. Some reports from the panels are expected in 1991. EEG believes this exercise is worthwhile and may lead to useful information. Both the ACNW and the NAS-WIPP Panel have expressed concern about whether there is enough technical input to the panels.

CONCLUSIONS

The human intrusion scenario will probably be the most difficult portion of the Standard for most repositories to meet. It should be retained because of its incentive to avoid mineral rich areas and to require a re-evaluation of the need for engineered enhancements and for a better understanding of the repository horizon. It is appropriate to use a distribution of values for drilling rates and actions by drillers but the distribution needs to be determined by a scientific consensus that includes "outside" technical experts.

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Section 6 - Performance Assessment

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 19, 1991

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: THE CONSISTENT USE OF PROBABILISTIC RISK ASSESSMENT

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, and in earlier meetings, we discussed the unevenness and inconsistency in the use of probabilistic risk assessment (PRA) in NRC. PRA can be a valuable tool for judging the quality of regulation, and for helping to ensure the optimal use of regulatory and industry resources, so we would have liked to see a deeper and more deliberate integration of the methodology into the NRC activities. Our recommendations to this end are directed at problems that took time to develop, and are likely to take a long time to solve.

PRA is not a simple subject, so there are wide variations in the sophistication with which it is used by the various elements of NRC. There are only a few staff members expert in some of the unfamiliar disciplines -- especially statistics -- that go into a PRA, so it is not surprising that there are inconsistencies in the application of the methodology to regulatory problems.

To illustrate the problems, let us just list a few of the fundamental aspects of the use of PRA, in which different elements of the staff seem to go their own ways. These are just illustrations, but each can lead to an erroneous regulatory decision.

1. The proper use of significant figures is in principle a trivial matter, but it does provide a measure of a person's understanding of the limitations of an analysis. Yet we often hear from members of the staff who quote core-damage probabilities to three significant figures, and who appear to believe that the numbers are meaningful. It is a rare PRA in which even the first significant figure should be regarded as sufficiently accurate to play an important role in a regulatory decision, but there is something mesmerizing about numbers, which imbues them with misleading verisimilitude.

   They deserve respect, but not too much, and it is wrong to err in either direction.

2. Closely related is uncertainty. There is no way to know how seriously to take the results of a PRA without some estimate of the uncertainty, yet we often hear thoroughly unsatisfactory answers (some perhaps invented on the spot) when we ask about uncertainty. One of the advantages of PRA is that it provides a mechanism for estimating uncertainty, uncertainty which is equally present, but not quantified, in deterministic analyses.

3. Conservatism. A PRA should be done realistically. The proper time to add an appropriate measure of conservatism is when its results are used in the regulatory process. If the PRA itself is done with conservative assumptions (more the rule than the exception at NRC), and is then used in a conservative regulatory decision-making process, self-deception can result, or resources can be squandered.

The inconsistent use of conservatism was illustrated by a pair of briefings at our April 1991 meeting, which included updates on proposed rules on license renewal and on maintenance. In the former case, we were told that a licensee could use PRA to add an item for later review, but never to remove one -- a one-way sieve. In the latter case we were told that PRA could be used to justify either enhancement or relaxation of maintenance requirements. Foolish consistency may be a hobgoblin, as Emerson said, but there is nothing foolish in seeking consistency in regulation.

4. The bottom line. It has been widely recognized since WASH-1400 that the bottom-line probabilities (of either core melt or immediate or delayed fatalities) are among the weakest results of a PRA, subject to the greatest uncertainties. (That doesn't mean they are useless, only that they should be used with caution and sophistication.) Yet we find staff members unaware of these subtleties, often dealing with small problems, justifying their actions in terms of the bottom-line probabilities. This is only in part due to the Backfit Rule, which almost requires such behavior; it is also inexperience and lack of sensitivity to the limitations of the methodology.

A number of staff actions and proposals use bottom-line results of a PRA as thresholds for decision making, often with the standard litany about the uncertainty in the reliability of these results. In fact, the quantified uncertainty in the bottom-line results of a PRA is just as important a number as the probability itself. It would be straightforward to employ a decision-making algorithm that prescribes a confidence level for the decision, and uses both the bottom-line probability and the uncertainty to achieve this. A further improvement would be to incorporate the consequences of erroneous decisions, what statisticians would call the loss function, into the decision-making process. The Commission has come close to this approach in its recent instructions to the staff on the diesel generator reliability question.

These are just a few examples of problems with the use of PRA in NRC, all common enough to be disturbing, and increasing in frequency as the use of PRA increases. It has been more than fifteen years since the publication of WASH-1400, a pioneering study which, despite known shortcomings, established the NRC at the forefront of quantitative risk assessment. One could have hoped that by now a coherent policy on the appropriate use of PRA within the agency, on both large and small problems, could have evolved.

We recommend that:

A. A mechanism be found (perhaps a retreat) through which the few PRA and statistical experts now scattered throughout the agency (and generally ignored) can be brought together with the appropriate senior managers and outside experts, to work toward a consistent position on the use of PRA at NRC. It could be worth the time expended. (Among other long-term benefits, such an interaction would add an element of horizontal structure to the NRC's predominantly vertical organization.)

B. The Commission then find a way to give credence and force to that position.

C. The Commission emphasize recruitment of larger numbers of professionals expert in PRA and statistics.

D. The Commission consider some kind of mandate that any letter, order, issue resolution, etc., that contains or depends on a statistical analysis or PRA, be reviewed by one of the expert PRA or statistical groups.

We do not pretend that this is an easy problem. The solution involves not only a cultural shift, so that those few experts already at NRC have some impact, but also substantial enhancement of the staff capabilities. That will require incentives that only the Commission can supply. It is interesting that the Commission's Severe Accident Policy Statement, dated August 1985, stated that "within 18 months of the publication of this severe accident statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs...."

Additional comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins are presented below.

Sincerely,

David A. Ward
Chairman

Additional Comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins

We thoroughly endorse this letter, and regret only that the Committee chose to ignore the parallels between the PRA problems and those in a number of other newer technologies significant to nuclear safety. Recommendation C should have included mention of some of these -- electronics and computers, for example -- which are of increasing importance. Weaknesses in those areas also need correction. Computerized protection and control systems, in particular, require the kind of sophisticated review that NRC is in no position to provide.

RAW 5-9-91

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

April 29, 1991

Mr. Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Bernero:

SUBJECT: INDIVIDUAL AND COLLECTIVE DOSE LIMITS AND RADIONUCLIDE RELEASE LIMITS

The Advisory Committee on Nuclear Waste has been developing comments, thoughts, and suggestions relative to individual and collective dose limits and radionuclide release limits. Since we understand that your staff is reviewing these same topics, we wanted to share our thoughts with you. In formulating these comments, we have had discussions with a number of people, including members of the NRC staff and Committee consultants. The Committee also had the benefit of the documents listed.

Basic Definitions

As a basic philosophy, individual dose limits are used to place restrictions on the risk to individual members of the public due to operations at a nuclear facility. If the limits have been properly established and compliance is observed, a regulatory agency can be confident that the associated risk to individual members of the public is acceptable. Because the determination of the dose to individual members of the public is difficult, the International Commission on Radiological Protection (ICRP) has developed the concept of the "critical group" and recommends that it be used in assessing doses resulting from environmental releases. As defined by the ICRP, a critical group is a relatively homogeneous group of people whose location and living habits are such that they receive the highest doses as a result of radionuclide releases. The group may be real (in which case their actual habits may be known or predicted) or hypothetical (in which case their habits may be assumed, based on observations of similar groups).

The dose to individuals within the critical group is assumed to be that received by a typical member of the group. The purpose of this approach is to ensure that members of the public do not receive unacceptable exposures while, at the same time, ensuring that decisions on the acceptability of a practice are not prejudiced by a very small number of individuals with unusual habits.

9105074009

Additional

General Background Material

Workshop

Technical Basis

for

EPA HLW Disposal Criteria

September 24-26, 1991

If the number of people being exposed is large, the question often arises as to how to quantify the societal impact of the individual exposures. The collective dose concept was developed for expressing that impact in a quantitative manner and, as such, it is a numerical expression of the summed doses to a given population.

In many respects, placing limits on total radionuclide releases from a nuclear facility is comparable to placing a limit on its total societal impact. In other words, placing a limit on the quantity of a given radionuclide that can be released is equivalent to placing a limit on the total societal impact that the facility can exert. This was the basis used by the U.S. Environmental Protection Agency (EPA) in setting release limits for a high-level radioactive waste repository, and it relates directly to EPA's basic criterion that the number of health effects should not exceed 1,000 during the first 10,000 years.

Underlying Assumptions

Although it is generally accepted that the dose received by an individual is a reasonable expression of the associated risk, it is questionable whether the collective dose is a true measure of the societal impact of the aggregate of exposures to individual members of a population. Implicit in the concept of collective dose is the assumption that the linear hypothesis is correct, that is, that there is a linear (non-threshold) relationship between the total dose to a population group and the associated health impacts.

In many ways, application of the collective dose concept leads to a paradox. At high doses and high dose rates where the risk coefficients are best known, the concept of collective dose cannot be applied since the dose-response curve is nonlinear; at low doses and low dose rates where linearity between dose and the associated health effects is assumed to apply, the risk coefficients are far less certain. This leads to additional restrictions in the application of the collective dose concept, as follows:

- The exposed population must be well known with respect to size and possibly age, sex, and temporal distributions.
- The exposure pathways must be characterized for the population at risk.
- Individual contributions to the collective dose must consist only of doses to the whole body, or to specific organs or tissues for which stochastic risk coefficients are known.

In short, application of the collective dose concept requires detailed knowledge of the exposed population and the radiation doses to its members. The collective dose concept is valid for representing the collective risk only if both of these factors can

be described and quantified, and it should be used for risk assessments only if the associated uncertainties are sufficiently small that the calculated collective dose itself is within an acceptable range of uncertainty. In addition, it is important to note that a high individual risk to a small number of people is not necessarily the same as a low individual risk to a large number of people, even though the collective dose may be the same. For this reason, expressions of societal risk in terms of collective dose should always include detailed data not only on the number of people exposed, but also on the number of people receiving exposures within each dose range. Although collective dose can be used as a surrogate for societal risk, its interpretation requires care.

Truncation of Collective Dose Calculations

On a theoretical basis, there is no justification for excluding the application of the linear hypothesis to the evaluation and interpretation of the societal impact of low doses and low dose rates on population groups. This hypothesis, in fact, has been generally accepted by the scientific community, including organizations such as the National Council on Radiation Protection and Measurements (NCRP) and the ICRP, as a valid basis for estimating the stochastic risks associated with low doses of ionizing radiation. If one accepts this observation, calculations of collective doses should include the doses to all individuals within the population group, regardless of how small the associated doses and/or dose rates may be. At the same time, however, it is important to recognize that there may be cogent reasons for not including within collective dose calculations extremely low doses to individual members of a population group. Several approaches that have been proposed and/or applied to justify such omissions are discussed below.

Following the concept that certain risks to individual members of the population are negligible, the NCRP has recommended (under what it defines as the concept of a "Negligible Individual Risk Limit") that annual doses to individual members of the population that are less than 0.01 mSv (1 mrem) be excluded from collective dose calculations. In interpreting this recommendation, however, it is important to understand the underlying principle on which it was based. Informal discussions with representatives of the NCRP revealed that truncation in this case was considered to be acceptable from the standpoint of societal impact, because the burden on society represented by any additional cancers among people receiving exposures in this dose rate range would not necessitate any additional medical facilities. Another approach for truncation that has been informally suggested by representatives of the NCRP is that it might be permissible to discard a collective dose (calculated on the basis of extremely low dose rates to members of an exposed population) provided that the

associated collective dose would not be estimated to result in one additional cancer.

Variations in the dose rates from natural background radiation sources have been proposed as another basis on which to truncate collective dose calculations. The contribution to collective dose from natural sources is large relative to that from many artificial sources. Consequently, it is often difficult to measure in a meaningfully quantitative manner very low dose rates to individual members of the population that arise from artificial sources. Thus, although there may be no biological basis for excluding very low dose rates from collective dose calculations, there is justification for excluding them on a statistical basis because of the uncertainties in the associated calculations.

Determinations of Compliance With Standards

From the previous discussion, it follows that the establishment of limits on the concentration of individual radionuclides in various environmental media (e.g., air and water) is comparable to the establishment of dose limits for individual members of the population. Likewise, the placement of limits on total radionuclide releases from a nuclear facility is comparable to the establishment of limits on the associated permissible collective doses to the affected population. In terms of the determination of compliance with a set of standards, it is readily possible to measure the concentrations of individual radionuclides in various environmental media, and it is similarly possible to estimate the associated doses to individual members of the population. In contrast, estimates of the total releases of radionuclides from a nuclear facility would require not only knowledge of the concentrations of individual radionuclides in all environmental media, but also the determination of the rate of movement (transport) of each radionuclide (including the evaluation of site-specific pathways) within all such media from the facility to the accessible environment. Similar uncertainties would accompany estimates of the associated collective doses.

Summary

In summary, the Committee offers the following statements on the benefits of the application of various limits for determining the public health risks associated with nuclear operations.

1. Individual dose limits can be used to limit the risks to individual members of a population group.

2. Collective dose limits can be used to limit the societal impacts of doses to a large number of individuals. The accuracy of collective dose as a measure of societal risk, however, depends on the validity of the linear (non-threshold)

hypothesis in assessing the stochastic effects of ionizing radiation.

3. Collective dose calculations are representative of societal risk only if certain conditions are satisfied; namely, the exposed population is defined and characterized with respect to size, age, and sex; the distribution of doses to individual members of the population is within a limited range; the exposure pathways have been characterized for the population at risk; and individual contributions to the collective dose consist only of doses to the whole body, or to specific organs or tissues for which stochastic risk coefficients have been adopted.

4. Techniques for measuring the concentrations of individual radionuclides in various environmental media, and for estimating the associated dose rates to individual members of the population, are readily available, and compliance with such limits can be determined. In contrast, the measurements that would be required to determine the total releases of individual radionuclides from a nuclear facility and estimations of the associated collective dose to all offsite population groups would be difficult.

5. Given the general acceptance of the linear hypothesis, there is no biological basis on which to truncate calculations of collective doses. Nonetheless, regulators must recognize that estimates of dose rates from artificial radiation sources, that represent only a few percent of those from natural radiation sources, carry with them large uncertainties and relatively little aggregate risk. Such uncertainties may well serve as a basis for truncating collective dose calculations at very low dose rates without adverse impacts on estimates of the associated risks.

We trust that these comments will be helpful. We plan to review and comment on your report regarding this subject when it becomes available, consistent with the SRM dated April 18, 1991.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

References:

1. International Commission on Radiological Protection, "1990 Recommendations of the International Commission on Radiological Protection," Publication 60, Annals of the ICRP (1991).
2. National Council on Radiation Protection and Measurements, "Recommendations on Limits for Exposure to Ionizing Radiation," Report No. 91 (1987).
3. National Radiological Protection Board, "Radiological Protection Objectives for the Land-based Disposal of Solid Radioactive Wastes." Consultative Document. NRPB-M279, March 1991.
4. "Possibilities and Limits of the Application of the collective Dose." A Recommendation of the Radiological Protection Commission (SSK). Bundesanzciger, No. 126a, July 1985.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

June 27, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: RESPONSE TO QUESTIONS ACCOMPANYING WORKING DRAFT #3 OF THE EPA STANDARDS

Draft #3 of the proposed Environmental Protection Agency (EPA) Standards for the management and disposal of spent nuclear fuel, high-level and transuranic radioactive wastes includes six questions. With the thought that our comments would be helpful, we have prepared the following summary responses to each of these questions.

Question 1:

Two options are presented in Sections 191.03 and 191.14 pertaining to maximum exposures to individuals in the vicinity of waste management, storage and disposal facilities: a 25 millirems/year ede limit and a 10 millirems/year ede limit. Which is the more appropriate choice and why?

Response:

The question, as phrased, refers to "maximum" exposures to "individuals." Because radionuclide releases from a high-level waste (HLW) repository, if they occur, could continue for a number of years, we have responded to the question in the sense of what would be the maximum acceptable annual exposure (dose) to members of the public over an extended period of time, in contrast to what might be considered an acceptable maximum exposure over a single year. This is in accord with the approach taken by both the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP).

In a similar manner, we assume that by maximum exposures to "individuals," the EPA means maximum exposures to a "critical population group," following the approach recommended by the ICRP. With those caveats, our response follows.

We believe an effective dose rate limit of 0.10 mSv (10 mrem) per year is more appropriate for several reasons:

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1. Recent evaluations indicate that the biological effects of ionizing radiation may be higher than previously estimated.

2. The population in question may be exposed to more than one radiation source.

3. A fraction of the current dose limit should be reserved for potential future radiation sources.

4. Radionuclide releases from a repository, if they occur, could continue over a long period.

Such a dose rate limit would also be consistent with the recommendations of international organizations such as the ICRP, the International Atomic Energy Agency, and as noted in the 1989 report prepared by the radiation protection and nuclear safety authorities of Denmark, Finland, Iceland, Norway and Sweden (commonly referred to as the "Nordic" Study).

Question 2:

A new assurance requirement is presented in Section 191.13 that would require a qualitative evaluation of expected releases from potential disposal systems over a 100,000-year timeframe. Are such evaluations likely to provide useful information in any future selecting of preferred disposal sites?

Response:

We recognize that the specification of the 10,000-year time limit is somewhat arbitrary. It is important that significant geologic or climatic changes do not occur in the near-term period following the 10,000-year limit. We also agree that many geologic and climatic events that may affect the evaluation of site performance can be meaningfully extended beyond 10,000 years. In these cases, such an extension could provide information that would be useful for comparing the relative merits of several potential repository sites. In general, however, and particularly in the evaluation of the merits of a single site, the uncertainties involved in such an extension would make the value of the associated assessments questionable. It is important to note that, although evaluations of site performance may be quantitative, the results are subject to interpretation.

Question 3:

Two options are presented in Section 191.14 and 191.23 pertaining to the length of time over which the individual and ground water protection requirements would apply: a 1,000-year duration and a 10,000-year duration. Which is the more appropriate timeframe and why?

Response:

Title 10 Part 60 of the NRC regulations specifies that containment of the radionuclides within the waste be substantially complete for a period not less than 300 years nor more than 1,000 years. This constraint, coupled with other requirements, including the stipulation that the groundwater travel time to the accessible environment be at least 1,000 years, is designed to ensure that protection of the individual and the groundwater will extend well beyond 1,000 years.

When one also considers the fact that, after only a few thousand years of decay, the health hazards of the high-level wastes will be no greater than that of the original unmined uranium ore, it becomes readily apparent that it should be possible to ensure individual and groundwater protection for a duration of 10,000 years. We therefore endorse the extension of this time period. Such an extension would also make this requirement compatible with the limitation on health effects resulting from an HLW repository.

Question 4:

In Subpart C the Agency proposes to prevent degradation of "underground sources of drinking water" beyond the concentrations found in 40 CFR Part 141 -- the National Primary Drinking Water Regulations. The Agency is aware, however, that there may be some types of ground waters that warrant additional protection because they are of unusually high value or are more susceptible to contamination. Should the Agency develop no-degradation requirements for especially valuable ground waters? If so, what types of ground waters warrant this extra level of protection?

Response:

We agree that pollution of "underground sources of drinking water" should not be permitted beyond the limits specified in the National Primary Drinking Water Regulations. We believe that a no-degradation requirement for certain large volume aquifers, that represent major long-term existing or potential drinking water sources, may represent undue stringency. A preferred approach would be to reject as potential sites for the storage or disposal of high-level radioactive wastes those land areas which, if contaminated, could have the potential for polluting such aquifers. However, the volume and present value of an aquifer should not be the sole criteria for identifying those that should be protected. Other criteria may become significant with the passage of time.

At the same time, we believe it is important to recognize that the dose rate from underground sources of drinking water, even if contaminated to the limits specified in the National Primary Drinking Water Regulations, would still contribute only a small

fraction (4 percent) of the current long-term dose rate limit for members of the public. Even considering the more restrictive limit for an HLW repository (as suggested in our response to Question 1 above), groundwater complying with the Drinking Water Regulations would contribute no more than 40 percent of the dose rate limit. In this sense, application of the Drinking Water Regulations to a repository represents a degree of stringency, especially because the primary pathway for public exposures from such facilities is through drinking water.

Question 5:

Two options are presented in Notes 1(d) and (e) of Appendix B pertaining to the transuranic waste unit: a 1,000,000 curies option and a 3,000,000 curies option. Which is the more appropriate TRU waste unit and why?

Response:

The number of curies of transuranic waste that would be comparable to 1,000 MTHM of spent fuel ranges from 1 to 6 million curies, depending on when the assessment is made. Accordingly, we believe that it would be reasonable to adopt the 3 million curie option.

Question 6:

The Agency is investigating the impacts of gaseous radionuclide releases from radioactive waste disposal systems and whether, in light of these releases, changes to the Standards are appropriate. To assist us in this effort, we would appreciate any information pertaining to gaseous release source terms, chemical forms, rates, retardation factors, mitigation techniques and any other relevant technical information.

Response:

Two reports that may be helpful are

1. W. B. Light, et al., "C-14 Release and Transport from a Nuclear Waste Repository in an Unsaturated Medium," Lawrence Berkeley Laboratory, Report LBL-28923 (June 1990).

2. W. B. Light, et al., "Transport of Gaseous C-14 from a Repository in Unsaturated Rock," Lawrence Berkeley Laboratory, Report LBL-29744 (September 1990).

In commenting on this subject previously, we have noted the following:

a. The total inventory of carbon-14 in a repository containing 100,000 MTHM is estimated to be about 100,000

curies. This compares to a global production of carbon-14 by cosmic radiation of 20,000 curies per year, a global inventory of about 230 million curies, and an atmospheric inventory of 4 million curies. In fact, release of all of the carbon-14 inventory in a repository would increase the atmospheric inventory by only about 2 percent; this compares to natural variations in the atmospheric inventory of 10 percent to 40 percent.

b. Based on an assumed inventory of 100,000 MTHM, the rate of release of carbon-14 from a repository that would be permissible under the existing EPA Standards would be about 1 curie per year. Experience shows that any carbon-14 that is released would rapidly mix in the atmosphere, and estimates are that the accompanying dose rate to a person on top of Yucca Mountain would be far less than 0.01 mSv (1 mrem) per year. We also note that the limit on the release rate of 1 curie per year for a repository compares to an average release rate of 10 curies per year from a typical 1,000 MWe light-water reactor.

At the time the EPA Standards were developed, considerations were limited to evaluations of a saturated site. In such a case, water transport and geochemical barriers would have been strongly influential in retaining the carbon-14. Subsequent consideration of Yucca Mountain (an unsaturated site) makes the existing EPA Standards inappropriate. We believe the limit for carbon-14 as specified in the proposed Standards should be relaxed. For additional discussion on this topic, we refer you to the transcript and minutes of the Advisory Committee on Nuclear Waste Working Group meeting held on March 19, 1991.

We trust these comments will be helpful. If appropriate, we request that you forward them to Mr. Floyd L. Galpin of the U.S. Environmental Protection Agency.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

Reference

EPA, 40 CFR 191 - Draft Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, dated April 26, 1991, with attachments.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

May 30, 1991

Mr. Robert M. Bernero, Director
Office of Nuclear Material
Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Bernero:

SUBJECT: REVIEW OF DRAFT SECY PAPER ON DEALING WITH UNCERTAINTIES

The Advisory Committee on Nuclear Waste has reviewed a copy of the draft SECY paper, "Staff's Approach for Dealing with Uncertainties in Implementing the EPA High-Level Waste Standards." As you know, the NRC staff is writing this paper in response to a request from the Commission for an explanation of the management of uncertainties during the process of evaluating compliance of a proposed repository with the probabilistic standards of the U.S. Environmental Protection Agency (EPA). During our 30th meeting, we had extensive discussions on this topic with members of your staff. The transcript of our April 24, 1991 meeting contains details of our comments and concerns.

The draft SECY paper and its accompanying document provide a broad view of the uncertainties that will need to be addressed during site characterization and the subsequent licensing process. Although the draft SECY paper includes discussion of methods to reduce uncertainties, we believe the staff has insufficiently clarified its role in the management of uncertainties that will remain after a license application is submitted. The draft SECY paper is also substantially silent on (1) the general program plan envisioned by the NRC staff for managing uncertainties, (2) the way in which rulemaking and similar protocols will be used to manage uncertainties that are likely to become important at the time of license hearings, and (3) the distinction between the role of the NRC and that of the U.S. Department of Energy in reducing and managing technical uncertainties. At the same time, the draft SECY paper includes extensive coverage of topics that could be interpreted as not being pertinent to the questions that need to be addressed. One example is the discussion of the benefits to be derived from the existing version of the EPA Standards. The discussion of collective versus individual dose limits should also be removed from the SECY paper.

Although the draft paper is partially responsive to the request of the Commission for a discussion of the management of uncertainties, there is a need to develop a program plan that (1) establishes guidelines for developing responses to a broad range of uncertainty

issues; (2) describes the bases for actions by the staff, for example, the method of balancing reliability and risk; and (3) serves as a guide to the preparation of additional reports that systematically explore the application of the overall plan to various parts of the licensing process, such as the approach to reconciling expert judgments that conflict. Such a plan would provide assurance of long-term regulatory consistency and completeness; in essence, it would serve as a "road map." The existing draft paper and our discussions with the NRC staff can readily serve as a beginning for the preparation of a program plan.

We believe that the staff is approaching the difficult and complex topic of uncertainty issues with growing insight. Although the present draft SECY paper represents an improvement over the earlier version, it demonstrates the need to organize the variety of issues to be addressed so that uncertainties are minimized and managed satisfactorily, leading to the formulation of defensible policies. Some parts of the draft paper, particularly portions of section 2 and much of section 3, could, after revision, be issued as a partial response to the Commission's request.

We look forward to working with the staff on these matters and to reviewing additional documents on this important topic as they are developed. We would be pleased to meet with you to elaborate on our comments and suggestions.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

Reference:
Draft SECY Paper, "Staff's Approach for Dealing with Uncertainties in Implementing the EPA High-Level Waste Standards," undated, received April 26, 1991.

STAFF POSITION 60-001

SUBJECT: CLARIFICATION OF THE 300-1000 YEARS PERIOD FOR SUBSTANTIALLY COMPLETE CONTAINMENT OF HIGH-LEVEL WASTES WITHIN THE WASTE PACKAGES UNDER 10 CFR 60.113(a)(1)(ii)(A)

THE QUESTION: Under the applicable performance objective in 10 CFR Part 60, may waste packages for high-level waste be designed for a lifetime in excess of 1000 years and, if so, may containment over the entire design lifetime be factored into required engineered barrier system and overall repository system performance assessments?

STAFF POSITION: The requirement in 10 CFR 60.113(a)(1)(ii)(A) for substantially complete containment of high-level wastes within the waste packages for a period not less than 300 years nor more than 1000 years following repository closure is a minimum performance requirement which is not intended, and should not be interpreted, as a cap on the waste package lifetime or a limitation on the credit that can be taken (in engineered barrier system and overall repository system performance assessments) if the waste package is designed to provide containment in excess of 1000 years.

ISSUANCE DATE: July 27, 1990

DISCUSSION: § 60.113 contains the subsystem performance requirements for both the engineered and natural barriers of the geologic repository. Specifically, 10 CFR 60.113(a)(1)(ii)(A) states that, assuming anticipated processes and events:

> Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account the factors specified in § 60.113(b) provided, that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository;

and as referenced, 10 CFR 60.113(b) states that:

> On a case-by-case basis, the Commission may approve or specify some other radionuclide release rate, designed containment period or pre-waste-emplacement groundwater travel time, provided that the overall system performance objective, as it relates to anticipated processes and events, is satisfied. Among the factors that the Commission may take into account are:
>
> (1) Any generally applicable environmental standard for radioactivity established by the Environmental Protection Agency;
>
> (2) The age and nature of the waste, and the design of the underground facility, particularly as these factors bear upon the time during which the thermal pulse is dominated by the decay heat from the fission products;
>
> (3) The geochemical characteristics of the host rock, surrounding strata and groundwater; and
>
> (4) Particular sources of uncertainty in predicting the performance of the geologic repository.

The phrase "not less than 300 years nor more than 1000 years" from § 60.113(a)(1)(ii)(A) can possibly be taken out of context and interpreted to mean (1) that the waste package must be designed to have a lifetime no greater than the stated period or (2) that, in assessing the performance of the waste package and the engineered barrier system, one must assume that the waste package fails at the end of the stated period. These interpretations would mischaracterize the "containment" requirement. Sound safety policy (as reflected in the rules) should encourage good waste package design, including a long period of expected containment. Either of the interpretations mentioned above - a limitation on the waste package lifetime or limitation on the period

for which containment may be factored into analyses - would be at odds with this desirable safety practice. Neither the language nor the regulatory history of the rule requires or supports any such interpretation. The waste package may be designed for a longer lifetime and such longer lifetime may be considered in evaluations of compliance with the engineered barrier system and overall repository system performance objectives.

I. Evolution of the "Containment" Rule

An understanding of the Commission's intent in the "containment" requirement of 10 CFR 60.113 is best obtained by a review of the rule from its development in proposed form to its promulgation in final form. The "containment" rule as proposed states in part that "... the waste packages will contain all radionuclides for at least the first 1000 years after permanent closure." (46 FR 35280, July 8, 1981). It is important to note that the containment period in the proposed rule was fixed as a single durational figure (i.e., 1000 years) and was unequivocally expressed as a minimum. However, a number of commentors expressed concern with the formulation of the proposed "containment" rule as well as the other subsystem performance requirements. They pointed out that these requirements were supposed to contribute to ensuring compliance with an overall EPA standard, yet, at the time the requirements were proposed and commented upon, there was no such EPA standard. (All that was available was an EPA "working draft," not even a proposed EPA rule.) The staff accordingly sought Commission guidance on the question whether to proceed with the numerical subsystem performance objectives as part of the final rule or to defer their publication until after the EPA standard had been issued. (SECY-82-427, "Commission Options on Developing Final Technical Criteria for Disposal of High-Level Waste in Geologic Repositories.") The language which the staff suggested for publication -- should the Commission approve -- would have required a specific containment period (1000 years), subject to adjustment to take into consideration a variety of factors, including the standard that EPA might actually promulgate. The specific text

(submitted as part of SECY-82-288, "10 CFR Part 60 - Disposal of High-Level Radioactive Wastes In Geologic Repositories: Technical Criteria") reads as follows:

> Containment of HLW within the waste packages will be substantially complete for a period of 1,000 years after permanent closure of the geologic repository, or such other period as may be approved or specified by the Commission.

The Commission decided that the staff should proceed to finalize the technical criteria, including numerical performance objectives for the waste packages. However, the Commission decided that the wording should be modified so that, as the final rule states, the containment period, to be determined by the Commission, "shall be not less than 300 years nor more than 1,000 years after permanent closure."

The change can be traced to a Commissioner's recommendation, dated December 8, 1982 (captioned "HIGH LEVEL WASTE TECHNICAL RULE"):

> Replacing the staff formulation of the designed waste package containment period in Section 60.113(a)(1)(ii)(A) (1,000 years after closure or such other period as may be approved or specified by the Commission) with the requirement that the Commission specify the appropriate period within a range of from 200 to 1,000 years, taking into account the four factors in Section 60.113(b). This should accomplish essentially the same purpose as the staff's formulation in a more neutral form.

(In subsequent Commission direction to the staff, the 200-year figure was changed to 300 years.)

This history establishes clearly that the Commission intended no departure from the principle that a specific minimum containment period for the waste package should be specified. This was a cornerstone of the proposed rule and

the staff's suggested revision, and the Commission undertook no change in that purpose. The only alteration was one designed to eliminate the apparent presumption that the minimum containment period was to be 1000 years; by providing a range, the length of this minimum period would be formulated "in a more neutral form." That is, the flexibility provided in the rule (considering factors specific to particular sites and designs) could be applied to set an appropriate minimum containment period; and so long as it was of sufficient duration to cover the period when radiation and thermal conditions in the engineered barrier system are dominated by fission product decay (Section 60.113(a)(1)(i)(A)), the rule expressed no further preference for any particular number of years within the range.

The Commission's views were developed in the statement of considerations accompanying publication of final technical criteria. The concern that was being addressed involved uncertainties arising out of thermal disturbances of the area near the emplaced waste: the specification of a minimum containment period (i.e., a prescribed period to be determined within a broad range) would limit the source term (i.e., radionuclide releases from the waste package) during the thermal pulse and thereby reduce these uncertainties. The discussion (48 FR 28194, June 21, 1983 at 28196) includes the following:

> ...the Commission continues to be concerned that thermal disturbances of the area near the emplaced waste add significantly to the uncertainties in the calculation of the transport of radionuclides through the geologic environment. The proposed rule addressed this problem by providing that all radionuclides should be contained within the waste packages for a period of 1,000 years. The Commission continues to consider it important to limit the source term by specifying a containment period (as well as a release rate). But the uncertainties associated with the thermal pulse will be affected by a number of factors, such as the age and nature of the waste and the design of the underground facility. For some repositories, a period substantially shorter than 1,000 years may be sufficient to allow for some of the principal sources of uncertainty

to be eliminated from the evaluation of repository performance. For cases analyzed by the Commission on the basis of specified assumptions, a range of 300 years to 1,000 years would be appropriate. (These values appear in § 60.113(a)(1)(ii)(A)). Yet even a shorter designed containment period might be specified, pursuant to § 60.113(b), in the light of conditions that are materially different from those that had been assumed. For example, if the wastes had been processed to remove the principal heat-generating radionuclides (cesium-137 and strontium-90), the 300 - years provisions would not be controlling.

Given this discussion, it is evident in the public record as well as internal documents that the Commission had in mind the fixing of a particular minimum containment period (generally within the 300 - 1,000 year range) that would suffice to eliminate some of the principal sources of uncertainty. (This is emphasized by the Commission's explanation, also at 48 FR 28196, of its defense in depth approach as one that would prescribe "minimum performance standards for each of the major elements of the repository.") There is nothing to suggest that the 300 - 1,000 year range would play any other part in the application of the requirements of 10 CFR Part 60.

II. Relation of the Containment Requirement to Other Post Closure Performance Objectives

As discussed below, the containment requirement has an intended relationship to both the overall repository system performance objective and the controlled release performance objective of the engineered barrier system.

A. Overall System Performance Objective

As noted above, the containment requirement was established as a measure that would limit the uncertainties arising out of thermal disturbances of the area near the emplaced waste. The underlying reason for limiting the uncertainties was the Commission's expectation that application of the EPA standard would be facilitated thereby. Thus, satisfaction of the containment requirement would

"ordinarily contribute to meeting the [EPA] standards"; and the "definite contribution" of the waste package would be important "for the Commission to be able to conclude that the EPA standard will be met." (48 FR at 28196.) Clearly, if the performance of the waste package in accordance with Section 60.113(a) is deemed to contribute to a finding of compliance with the EPA standard, then the containment for an even longer period should make even more of a contribution. To the extent warranted by the data submitted in support of the license application, containment of radionuclides within the waste packages can and should be recognized in applying the EPA standard, without any arbitrary time limitation.

B. Controlled-Release Performance Objective

The close relationship between the two performance objectives in 10 CFR 60.113(a)(1)-- the containment requirement and the controlled-release requirement-- needs to be recognized (See 48 FR 28209). These are coupled requirements that serve to control the release of radionuclides to the geologic setting and thereby contribute to meeting the EPA standard. The controlled-release performance objective specifies that following the containment period, the release rate of any radionuclide from the "engineered barrier system" shall not exceed specified values. The issue arises because the Commission indicated its intention that each of the multiple barriers described in the rule must make a definite contribution to satisfying the EPA standard; a valid question, then, is whether credit for containment of radionuclides in the waste package beyond the containment period can be taken in judging whether satisfaction of the controlled-release performance objective would in fact contribute to meeting the EPA standard. The text of the rule is really unambiguous on the point, as the controlled-release requirement pertains to the "engineered barrier system," which by definition includes the waste packages. Accordingly, in determining whether releases from the engineered barrier system are kept low enough, one must consider the role of the components of that system, and that includes the role of the waste packages. Once again, whatever degree of containment can be demonstrated by the applicant will be recognized in determining whether the engineered barrier system is making the contribution envisaged by the Commission.

III. Conclusion

Inasmuch as the waste package will be assessed by the NRC staff on the merits of its design and its anticipated performance in the repository setting, the staff can give credit, if warranted, for waste packages designed to provide containment in excess of 1000 years. In other words, the staff would not arbitrarily assume in its compliance assessment for the waste package and engineered barrier system that the waste package will fail at 1000 years. The staff recognizes the licensee's option to do more than just meet the Commission's requirements (i.e., the minimum standards) and that, when warranted, the staff's assessments should reflect those design enhancements. In this regard, the purpose of the natural and engineered barriers subsystem requirements is to add confidence that the overall EPA containment requirements will be met. One way of minimizing uncertainties related to compliance with the EPA standard is to propose a waste package design for containment well in excess of 1000 years and the DOE could factor this design into the performance assessment which will be documented in its license application.

For the reasons cited above, the 300 - 1000 year containment period specified in 10 CFR 60.113(a)(1)(ii)(A) is not to be viewed as the waste package lifetime but rather the minimum period for which substantially complete containment of radionuclides within the waste package must be provided.

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