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Waste Management

Nuclear Waste Management and Transportation Quarterly Progress Report October Through December 1975

March 1976

**Prepared for the Energy Research
and Development Administration
under Contract E(45-1):1830**

Battelle
Pacific Northwest Laboratories

BNWL-1978

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PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
Under Contract E(45-1)-1830

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy ~~\$4.50~~; Microfiche \$2.25

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NUCLEAR WASTE MANAGEMENT AND
TRANSPORTATION QUARTERLY PROGRESS
REPORT OCTOBER THROUGH DECEMBER 1975

Compiled by

A. M. Platt
Nuclear Waste Technology Program

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March 1976

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FOREWORD

This document is one of a series of technical progress reports designed to report on all Division of Waste Management and Transportation (DWMT) funded programs at Battelle, Pacific Northwest Laboratories, with the exception of the Commercial Radioactive Waste Fixation Program. These reports are issued on a quarterly basis.

The prior report in this series is Nuclear Waste Management and Transportation Quarterly Progress Report July through September 1975, BNWL-1952.

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1.0 ADVANCED METHODS FOR MANAGEMENT AND DISPOSAL OF RADIOACTIVE WASTES

J. W. Bartlett - Project Manager

The objectives of this program are to 1) develop comprehensive plans for a program to select and implement methods for disposal of radioactive wastes and 2) extend the base for the plans and program via selected topical studies.

SUMMARY

Work in this program has been re-structured to reflect ERDA priorities related to final storage and disposal of nuclear wastes and assessment of alternatives for closing the back end of the LWR fuel cycle.

A model to describe diffusion-controlled migration of radionuclides from a repository in geologic formations was developed; a computational scheme is now being prepared. Models of migration in faulted media were also developed and found to differ from models for homogeneous media only in numerical values for coefficients.

The report on surveys of public attitudes toward waste management alternatives was completed and issued for internal review.

TASK 1. DEVELOPMENT OF ADVANCED WASTE MANAGEMENT PLANS -

J. W. Bartlett

The objective for this task is to prepare plans for R&D programs on storage and disposal of commercial radioactive wastes. During this quarter work focused on revising program structure and activities in accordance with requirements of ERDA's new organizations and priorities. Results of these revisions are outlined below.

Background

The AWM program originated in FY 1972 as an overview planning, concept development, and systems analysis function of the Division of Waste Management and Transportation. It has produced numerous reports and white papers on waste management concepts and issues. These reports have been used as basis for waste management decisions and formulation of R&D programs. A report on high-level waste disposal alternatives (BNWL-1900) has become a worldwide standard reference; it was the basis for current ERDA programs on geologic and seabed disposal. The work on nuclide migration is currently at the forefront of world effort. No other nation has an effort soliciting information on public attitudes toward risk and other factors involved in waste management systems; this work has evoked a high level of international interest.

Under the former DWMT, this program provided a quick-response resource of expertise to review and evaluate proposed nuclear waste management activities. It also served as a principal in communications with the press, public, and technical communities.

The AWM program previously contained three tasks: development of R&D plans, development of evaluation methods, and special studies. Plans for future work are structured to reflect two major ERDA interests: final storage/disposal of nuclear waste and waste management technology and systems (other than final storage/disposal) to close the back-end of the LWR fuel cycle. This restructuring is based on progress from BNWL-1900, BNWL-1904 (DRAFT), and recent experience related to producing the Technical Alternatives Document (TAD).

PLANS FOR FUTURE WORK

TASK A. FINAL STORAGE/DISPOSAL OF NUCLEAR WASTE

Objectives: This task will provide:

- Overview and systematic methods for establishing several acceptable final storage repositories in the contiguous U.S.
- Continuing assessment of alternative concepts for disposal.
- Continuing service as a principal in communication with the press, public, and technical communities.

Activities in this task include:

A-1. Final Storage

To FY 1977, this program element is focused on three efforts:

- Development of a methodical plan to provide, with reasonable assurance of success, several acceptable final geologic storage sites;
- Assessment of criteria and alternative technologies to implement the plan, and
- Assessment of the implications and constraints of retrievability.

A-2. Alternative Disposal Concepts

Disposal implies no commitment to retrievability. It permits use of geologic isolation concepts such as rock-melting, deep well injection, and solution-mined cavities. It also may be attained by converting storage to disposal.

All storage and disposal concepts involve reliance on barriers (manmade and natural) to maintain long-term isolation of waste from the environment and man. The basic barriers are waste form, engineered features such as encapsulation and geospheric retention of nuclides. In disposal, only geologic stability and nuclide retention are used as barriers.

Work in this task will develop and apply methods to evaluate barriers in alternative disposal systems. One part of the work is projection of nuclide migration, discussed below as a separate activity because of the scope of work involved.

Work on the systems aspect of barriers associated with disposal is scheduled to begin in the second half of FY 1976.

A-3. Nuclide Migration

Under AWM, PNL recently completed development of conceptual nuclide transport models for the geologic media being considered for final geologic storage. The models have not been tested or applied because data are not available.

Major issues to be resolved include:

Models

- Validity of assumptions
- Nuclides considered
- Accuracy required

Pathways considered

Data requirements

Inclusion of thermodynamic and kinetic effects

Validation of results of use

Data

Validity and methods for lab data

Feasibility and methods for field data

Inhomogeneity of nuclide species and geologic media

Scope and type of data required

Use of existing sources (Gabon, explosion cavities, etc.)

A detailed work plan for activities in this sector is currently being prepared. A workshop on data requirements is planned for February 1976, and planning for a companion workshop on models is under way. These workshops will define program work requirements consistent with the "reasonable assurance of natural protection" philosophy.

A-4. Public Attitudes Toward Storage and Disposal Alternatives

The recently completed study of public attitudes toward waste management alternatives indicated some substantial differences between nuclear technologists and other public groups with respect to factors such as acceptable levels of risk and relative importance of various risk factors. Such results can aid selection of publicly acceptable waste management methods from among alternatives available.

Past work is being usefully extended to aid decisions concerning alternatives by work in two areas: comparison of perceived and calculated risk, and quantification of social factors involved in site selection. A preliminary effort to

develop methods for the perceived and calculated risk comparison is already under way.

TASK B. ASSESSMENT OF ALTERNATIVE FUEL CYCLE AND WASTE MANAGEMENT

SYSTEMS

Objectives: This task will provide:

- Continuing compilation and description of viable alternative operations, concepts, and technologies for the post-fission part of the LWR fuel cycle, and
- Analytical methods and results for assessment of alternative systems.

The assessment analyses are expected to be useful for selecting preferred systems from the alternatives available on the basis of operational costs, risks, benefits, and potential environmental and logistical consequences.

Work on this task is currently inactive because of diversion of personnel to preparation of the Technical Alternatives Document (TAD). Preparation of TAD has greatly accelerated progress in this task since it will compile essential background information.

Work on TAD has produced the comprehensive generic concept of alternative systems shown in the attached Figure 1. This system concept is the basis for development of the analytical methods, which can be accomplished by adapting existing computer modeling techniques to this purpose.

Application of TAD to this task is expected to start early in CY 1976, with initial efforts focused on developing the analytical methods and review of the availability of technologies to form a complete system.

FIGURE 1. Generic Concept of Alternative Systems

TASK 2. DEVELOPMENT OF EVALUATION METHODOLOGY

Diffusion of Radionuclides from Geologic Formations -

H. C. Burkholder and C. Defigh-Price

To evaluate the risks from the underground storage of nuclear waste materials, mathematical models are needed to determine the consequences of radioactivity releases by various pathways to man. A model has previously been developed to describe the transport of nuclides by groundwater following a leach incident at a repository in a non-salt geologic formation and has been applied to an evaluation of the incentives for partitioning high-level waste.^(1,2)

The previous model assumes that the transporting medium is flowing water, but situations exist (e.g., a flooded emplacement shaft for a repository in monolithic granite or a seabed repository) for which water contacts the waste but moves very slowly or not at all. For these situations transport is predominantly by molecular diffusion in the liquid phase and cannot be described by the previous model, which becomes indeterminate when the water velocity becomes zero. In addition, there are situations where the waste becomes vaporized and diffuses through the medium in the gas phase (e.g., a repository in a salt formation where the plasticity of the medium permits the waste to move together and overheat or a repository which experiences unexpected volcanism). We have developed and solved a mathematical model which can be applied to such situations.

We make all of the previous model's assumptions⁽¹⁾ except that convective transport is neglected and the molecular diffusivities of the various chain members are assumed to be different. The latter assumption is unnecessary for liquid phase applications, where diffusivity varies only slightly for a wide range of migrating species but may be necessary for gas phase applications where diffusivities can vary significantly. The equations which describe the material balances of the "i" th chain member and all preceding chain members over a differential volume of the soil column are:

$$D_1 \frac{\partial^2 N_1}{\partial z^2} - K_1 \frac{\partial N_1}{\partial t} - K_1 \lambda_1 N_1 = 0$$

$$D_2 \frac{\partial^2 N_2}{\partial z^2} - K_2 \frac{\partial N_2}{\partial t} - K_2 \lambda_2 N_2 + K_1 \lambda_1 N_1 = 0$$

⋮

$$D_i \frac{\partial^2 N_i}{\partial z^2} - K_i \frac{\partial N_i}{\partial t} - K_i \lambda_i N_i + K_{i-1} \lambda_{i-1} N_{i-1} = 0$$

The boundary conditions for impulse and band releases are the same as in the previous model.⁽¹⁾ The model has been solved by the Laplace transform technique for both types of release. For illustration we consider an impulse release whose boundary conditions are described in Table 1. The analytical solution for a two-member chain is:

$$N_1 = N_1^0 \frac{D_1}{L^2} \frac{n}{2} \sqrt{\frac{K_1}{\pi D_1^3}} \exp\left(-\frac{K_1 n^2}{4\theta} - R_1 \theta\right)$$

$$N_2 = N_2^0 \frac{D_1}{L^2} \frac{n}{2} \sqrt{\frac{K_2 D_1}{\pi D_2^3}} \exp\left(-\frac{K_2 D_1 n^2}{4 D_2 \theta} - \frac{R_2 D_2}{D_1} \theta\right) + \frac{N_1^0 \frac{D_1}{L^2} K_1 R_1 \frac{D_1}{D_2}}{\left(K_1 - K_2 \frac{D_1}{D_2}\right)} \exp\left(-\frac{R_1 K_1 - R_2 K_2}{K_1 - K_2 \frac{D_1}{D_2}} \theta\right)$$

$$\left\{ \exp\left(-\sqrt{\frac{K_1 K_2 (R_2 - R_1 \frac{D_1}{D_2})}{K_1 - K_2 \frac{D_1}{D_2}}} n\right) \left[\operatorname{erfc}\left(\frac{n}{2} \sqrt{\frac{K_2 D_1}{D_2 \theta}} - \sqrt{\frac{K_1 (R_2 \frac{D_2}{D_1} - R_1)}{K_1 - K_2 \frac{D_1}{D_2}}} \theta\right) - \operatorname{erfc}\left(\frac{n}{2} \sqrt{\frac{K_1}{\theta}} - \sqrt{\frac{K_2 (R_2 - R_1 \frac{D_1}{D_2})}{K_1 - K_2 \frac{D_1}{D_2}}} \theta\right) \right] \right. \\ \left. + \exp\left(+\sqrt{\frac{K_1 K_2 (R_2 - R_1 \frac{D_1}{D_2})}{K_1 - K_2 \frac{D_1}{D_2}}} n\right) \left[\operatorname{erfc}\left(\frac{n}{2} \sqrt{\frac{K_2 D_1}{D_2 \theta}} + \sqrt{\frac{K_1 (R_2 \frac{D_2}{D_1} - R_1)}{K_1 - K_2 \frac{D_1}{D_2}}} \theta\right) - \operatorname{erfc}\left(\frac{n}{2} \sqrt{\frac{K_1}{\theta}} + \sqrt{\frac{K_2 (R_2 - R_1 \frac{D_1}{D_2})}{K_1 - K_2 \frac{D_1}{D_2}}} \theta\right) \right] \right\}$$

TABLE 1. Boundary Conditions for Impulse Release

Initial $t = 0, \text{ all } Z$	Inlet $t > 0, Z = 0$	Outlet $t > 0, Z = \infty$
$N_1 = 0$	$N_1 = N_1^0 \delta(t)$	$N_1 = \text{finite}$
$N_2 = 0$	$N_2 = N_2^0 \delta(t)$	$N_2 = \text{finite}$
$\vdots$	$\vdots$	$\vdots$
$N_i = 0$	$N_i = N_i^0 \delta(t)$	$N_i = \text{finite}$

$\delta(t)$ is the unit impulse function and N^0 is the inventory at the time of release

The general form of the analytical result shown here is quite similar to that of the previous model. Therefore, we can expect that the effects of diffusion, sorption, and radioactive decay will be similar to those of the previous model. However, because diffusion is the sole transport mechanism, nuclides should take much longer to enter the biosphere and

thus have lower activity levels. In contrast to the rather sharp distinct peaks from the previous model, we expect broad overlapping peaks from the present model.

During the next quarter a computational scheme similar to the GETOUT code⁽²⁾ will be developed and computations will be made to compare the nuclide discharge into the biosphere from this model with that of the previous model.

Nuclide Migration Models for Faulted Media - H. C. Burkholder

There are two extreme types of geologic media which can be termed "particulate" and "monolithic." (The possibility also exists for salt to be present or not present in each). Water moves through the interstices of the particulate media and through the cracks or faults in the monolithic media. We have previously modeled the particulate medium as a cylindrical column filled with particles

on which equilibrium sorption takes place.⁽¹⁾ If we consider a single nuclide and make a material balance over a differential column distance and a differential time, we have the following:

$$\frac{\partial c}{\partial t} + \left(\frac{K_d \rho}{\epsilon} \right) \frac{\partial c}{\partial t} + V \frac{\partial c}{\partial Z} = D_p \frac{\partial^2 c}{\partial Z^2} - \lambda c - \lambda \left(\frac{K_d \rho}{\epsilon} \right) c.$$

The first and second terms represent the accumulation of nuclide in the dissolved state and in the sorbed state. The third and fourth terms represent convection and dispersion in the dissolved state. The fifth and sixth terms represent radioactive decay in the dissolved state and in the sorbed state.

Rearranging, we have:

$$\left(1 + \frac{K_d \rho}{\epsilon} \right) \frac{\partial c}{\partial t} + V \frac{\partial c}{\partial Z} = D_p \frac{\partial^2 c}{\partial Z^2} - \lambda \left(1 + \frac{K_d \rho}{\epsilon} \right) c$$

and finally

$$K \frac{\partial c}{\partial t} + V \frac{\partial c}{\partial Z} = D_p \frac{\partial^2 c}{\partial Z^2} - \lambda K c \quad (1)$$

where:

c = nuclide concentration

Z = axial distance

t = time

V = water velocity

D_p = axial dispersion coefficient for the particulate medium

λ = decay constant

$$K = 1 + \frac{K_d \rho}{\epsilon} = \text{sorption equilibrium constant}$$

K_d = distribution coefficient

$$\left[\frac{\text{moles in sorbed state}}{\text{mass bed}} \right] \div \left[\frac{\text{moles in dissolved state}}{\text{volume of solution}} \right]$$

ρ = bed density $\left[\frac{\text{mass bed}}{\text{bed volume}} \right]$

ϵ = void fraction $\left[\frac{\text{volume of solution}}{\text{bed volume}} \right]$.

Monolithic media can be modeled on the same basis. The fault can be considered as a cylindrical open column with equilibrium sorption occurring on the walls. If we again consider a single nuclide and make a material balance over a differential column distance and a differential time we have the following:

$$\frac{\partial c}{\partial t} + K_a R_f \frac{\partial c}{\partial t} + V \frac{\partial c}{\partial Z} = D_f \frac{\partial^2 c}{\partial Z^2} - \lambda c - \lambda K_a R_f c.$$

The physical description of terms for this model is the same as that for the particulate model.

Rearranging, we have:

$$(1 + K_a R_f) \frac{\partial c}{\partial t} + V \frac{\partial c}{\partial Z} = D_f \frac{\partial^2 c}{\partial Z^2} - \lambda (1 + K_a R_f) c$$

and finally

$$K' \frac{\partial c}{\partial t} + V \frac{\partial c}{\partial Z} = D_f \frac{\partial^2 c}{\partial Z^2} - \lambda K' c \quad (2)$$

where:

D_f = axial dispersion coefficient for the monolithic medium

$K' = 1 + K_a R_f$ = sorption equilibrium constant

K_a = distribution coefficient $\left[\frac{\text{moles in sorbed state}}{\text{surface area}} \right] \left[\frac{\text{moles in dissolved state}}{\text{volume of solution}} \right]$

R_f = surface to volume ratio for the fault (assumed a cylinder in this case).

A comparison of the result of the two derivations (Equations 1 and 2) shows that their form is identical. The only differences are: 1) the numerical value of the dispersion coefficient and 2) the relationship for calculating the scheme sorption equilibrium constant. This result suggests that the computing scheme GETOUT is applicable to all media without salt, needing only properly determined values of the water velocity V , the dispersion coefficient D , and the sorption equilibrium constant K , for its application.

Public Attitudes Toward Nuclear Waste Management - W. S. Maynard (HARC)

The final report on FY 1975 activities was completed and issued for internal review. Results and conclusions are summarized below. Previous reports in these quarterlies outlined the scope and methods of the study.

Results

Sampling. Results are based upon the questionnaire responses from 465 study participants. The sample included 22 groups comprised of: junior high, high school, and university students; members of church, civic, service, and recreationist groups; members of environmental interest

groups; public utility employees; public officials and nuclear engineers and scientists. Discriminant analyses, based on the questionnaire responses to a set of attitude-belief statements and questions concerning respondents' reactions to the study, produced six major "clusters" of respondents. Survey results were presented using average responses by the total sample and by the six respondent clusters.

Priorities among Nuclear Waste Disposal Issues. Three different measurement methods (ranking, ratio estimation, and policy capturing/regression analysis) were used to assess the relative importance which respondents attached to four factors or issues associated with the design of nuclear waste disposal systems: Short-Term Safety, Long-Term Safety, Cost, and Accident Detection and Recovery. Although there were some subtle differences in the results produced by the three measurement methods, there was generally strong agreement among the three sets of results. For the total respondent sample, Long-Term Safety received the highest priority or weight, followed closely by Short-Term Safety and Accident Detection and Recovery. Cost received the lowest priority across the three measurement methods. Comparison among the respondent clusters indicated that Environmentalists clearly attached the greatest importance to Long-Term Safety, while Nuclear Technologists tended to place the greatest emphasis on Short-Term Safety considerations.

Evaluations of Levels of Risk and Cost. Respondents were asked to

judge the satisfactoriness of several levels of risk and cost which might be associated with nuclear waste disposal systems. For Short-Term Safety, fatality rates in excess of one death per year were generally seen as unsatisfactory by the total respondent sample. For Long-Term Safety, fatality rates exceeding one death in 10 years were judged unsatisfactory.

An increase of \$3.00 in the monthly household Cost for electricity were generally judged as satisfactory. For Accident Detection and Recovery, reductions in fatality rates of less than 50% were judged unsatisfactory. Comparisons among clusters of respondents indicated the following. Nuclear Technologists rated all levels of Short-Term and Long-Term risks as more satisfactory than did the other six respondent clusters. Environmentalists gave the lowest satisfaction ratings to levels of Short-Term and Long-Term risks. There were no significant differences among the clusters with respect to levels of Cost. All groups rated the lower levels of fatality reduction rates as unsatisfactory. Environmentalists were least satisfied with higher levels of fatality reduction, and Junior High/High School Students gave the highest satisfaction ratings to these higher levels.

Attitude-Belief Measurement. The six respondent clusters were compared according to their agreement or disagreement with five sets of attitudinal statements which addressed the following areas: the four nuclear waste disposal factors; severity of the nuclear waste problem; safety of nuclear power; need for nuclear power as well

as other sources of power; and efficacy of alternatives to nuclear power. For all five areas, the Nuclear Technologists and the Environmentalists were the most dissimilar in their attitudes and beliefs. Average responses for the other six clusters of respondents tended to fall between these two respondent groups. The attitude statements which evoked the greatest degree of consensus among respondents were those dealing with the four nuclear waste disposal factors. However, the Nuclear Technologists and Environmentalists differed considerably in their attitudes about the salience of Long-Term Safety.

Nuclear Technologists and Public Utility Employees, to a lesser degree, felt the problem of nuclear waste was less serious than did the other respondent groups. The Nuclear Technologists and Environmentalists were most dissimilar in their attitudes about the safety of nuclear power; average responses for the other groups tended to fall midway between the first two groups. There was general consensus among groups that more power will be needed in the future, and with the exception of the Environmentalists, nuclear power was perceived as a promising source of power. The Nuclear Technologists strongly disagreed that solar or geothermal power could solve the problem of additional power in the near-term, while the other respondent groups were less certain in their beliefs about these sources of power. Nuclear Technologists disagreed that conservation would remedy the need for more power, while Environmentalists were more optimistic about conservation.

Respondent Evaluation of Survey.

The majority of the respondents in the study agreed: that the public should be consulted on issues such as nuclear waste disposal; that the method used in this study (film/questionnaire) was an effective tool for obtaining input; and that the four factors used in the study were an effective way to conceptualize the problem. A majority of the respondents added their own comments about the study. Comments ranged from generally positive and supportive to very negative and critical of the study. Comments ranged from generally positive and supportive to very negative and critical of the study. Many respondents emphasized that the public needed more information in order to make informed judgements about nuclear waste disposal.

Conclusions

Conclusions derived from this study are structured around three questions:

1. What do the data tell us about social values associated with nuclear waste disposal and with related issues?
2. What are the potential policy implications of the results?
3. What is the methodological relevance of this research? What work remains to be done?

What do the data tell us about social values associated with nuclear waste disposal and with related issues?

With respect to the four waste disposal factors stressed in this study, perhaps one of the most salient findings was the relatively high importance which many respondents felt

should be placed on Long-Term Safety. The majority of the study participants felt that much attention should be placed upon the protection of future generations from the wastes being generated today.

One objective in sample selection was to survey public officials and nuclear scientists currently involved in nuclear technology research and planning, and to compare their perceptions and values with those of public groups whom they serve. The results of the importance measurements suggested that the Nuclear Technologist groups placed somewhat less relative weight on Long-Term Safety than they did on Short-Term Safety. The Nuclear Technologists did not downgrade the importance of Long-Term considerations by any means, but they placed somewhat less priority upon this issue than did those respondents representative of the general public, as well as respondents more critical of the nuclear industry.

The Environmentalist cluster clearly attached the greatest importance to Long-Term Safety issues. These groups did not downgrade the significance of Short-Term Safety but tended to relegate it to a slightly less important role when it was contrasted with Long-Term Safety. Indeed a dominant theme running through the results from several parts of the survey was the high priority placed on the Long-Term consideration by the Environmentalists. A major tenet in the environmentalist philosophy is the notion of man as a temporary steward of the environment, responsible for conserving and preserving it for the future. In this study, Long-Term

Safety was defined within the context of nuclear waste disposal; the results suggest that this factor was perceived as a component in the environmental ethic.

Another theme which emerges from the data is the polarization of values and attitudes among the various respondent groups. The Nuclear Technologists and Environmentalists, in particular, differed greatly on a number of key issues. Descriptions of the results repeatedly contrasted and commented upon these two groups' responses, sometimes to the relative exclusion of the other groups of respondents. In many instances the results showed Nuclear Technologists to be on one side of an issue, the Environmentalists clearly on the other side, with the remaining groups somewhere "in the middle." The sampling strategy was designed to incorporate groups whose views toward nuclear power differed greatly as well as groups whose orientations were less pronounced. The results from the measurement instruments used in the study suggest that this sampling objective was reasonably well met.

The finding that Cost was a relatively unimportant factor for most respondents also warrants mention. Care was taken in the way the four factors were operationally defined so that the expected values for the factors would be within the range of risks and costs for disposal systems. Given these best estimates of risk and cost, cost was judged to be of relatively little concern. While the data do not indicate that "cost is no

object," they do reveal that economic considerations pale when compared to fatality risks for the present generation or for future generations.

One section of the questionnaire was devoted to measuring the perceived satisfactoriness or acceptability of several levels of risk and cost. The levels which respondents were asked to judge bracketed the range of risks and costs expected to be associated with waste disposal systems. The data indicate that the respondent groups varied considerably in their tolerance for risk, but agreed fairly well with respect to the acceptability of different levels of cost. Nuclear Technologists and Environmentalists differed by as much as two or three orders of magnitude in the judged satisfactoriness of Short- and Long-Term risk levels. These two groups set the risk boundaries within which the average responses for the other respondent groups fell. For the factor, Accident Detection and Recovery, there was close agreement among the respondent groups that low levels or no reduction in fatality levels were unsatisfactory. At the higher levels of risk reduction (e.g., 90-95%), there was a divergence among the average responses. Although all groups judged these higher levels to be more satisfactory than the lower levels, the Environmentalist cluster was the least tolerant of less-than-100% reduction levels.

Clearly, the data suggest that the answer to the question, "How safe is 'safe enough'?" varies considerably

among individuals and among groups of the public. Using the aggregated responses for the total sample of respondents, some tentative "best estimates" were derived for what appear to be satisfactory levels of risk and cost for nuclear waste disposal systems. For Short-Term Safety, a satisfactory level of risk probably lies in the area of one death in 10 years and one death in 50 years. A satisfactory level of Long-Term Safety probably lies between one death in 100 years and one death per 1000 years. For Cost, a satisfactory level of increase in monthly household electricity is about \$3.00. Satisfactory levels of Accident Detection and Recovery were those approximating a 95% reduction in fatality rates.

While the respondent groups varied somewhat in assessment of the relative importance of nuclear waste disposal factors, the points of greatest polarization and variation in respondent values were on sets of issues of which nuclear waste disposal is one component. Nuclear Technologists and Environmentalists differed greatly in their perceptions of the severity of the nuclear waste problem and on their beliefs about the risks associated with nuclear power in general. These two groups also differed in the extent to which nuclear power was seen as an acceptable source of additional energy. There was some consensus that more power will be needed in the future, but much disagreement as to how these needs should be met. Respondents frequently mentioned that while nuclear waste disposal was a critical problem, their overriding question was whether nuclear power

should continue to be developed at its present rate.

Results from the final part of the questionnaire indicated that most respondents felt it a good idea to consult the public on technological problems such as nuclear waste disposal and generally agreed that the approach used in this study was a reasonable way to obtain public input. It is interesting to note that those groups who differed most from one another on value-attitude issues tended to be least supportive of the methodology, although the majority of the Nuclear Technologists and Environmentalists tended to support the film/questionnaire methodology. Frequent criticisms of the method were that it "failed to address the 'real' issues" or that it was overly simplistic in its treatment of the subject matter. The self-report data from respondents tends to indicate that the film/questionnaire method had some informational/educational value for most of the respondents. However, a frequent comment was that more information would be needed for them to deal adequately with value issues regarding nuclear waste disposal.

What are the potential policy implications of the results?

This study was designed to obtain public value information concerning nuclear waste disposal at a relatively early stage of overall planning and assessment activities. A number of specific findings may be relevant for planners and decision makers in early assessments of potential disposal options. First, Long-Term Safety was a high priority item

in the minds of numerous respondents. Second, respondents judged Cost to be of minor importance when contrasted with risk considerations. Third, some tentative estimates of acceptable levels of risk were identified. To the extent that disposal options considered vary in Short- and Long-Term Risk, Cost, and Accident Detection and Recovery, some of these findings may be useful in assessing these options from the standpoint of public acceptance.

Some of the more general findings also appear highly relevant from a policy perspective. For example, there were pronounced differences among respondent groups in their appraisal of acceptable risk levels, in the priorities they attached to the disposal factors, and in their values and attitudes toward broader issues surrounding the topic of nuclear waste disposal. There is fairly consistent evidence that respondents who were representative of planners and decision makers differed considerably on many key issues from respondents who were more representative of the general public (and particularly from respondents with strong reservations about the efficacy of nuclear power).

The data-based results and impressionistic information gathered by the survey administrators indicate that most respondents were willing to grapple with hard questions about risk and to make considered judgments about acceptable and unacceptable risk levels. Policy making about cost/benefit and risk/benefit questions has traditionally been based on expert judgment; much of the public

involvement in these policies has traditionally been reactive in nature, as evidenced by the consumerism movement, for example. In short, broad-based public input into risk/benefit decision making has been relatively lacking. Results presented in this report suggest that it is possible to obtain information about perceived risk from a broad range of the public.

Finally, the results which point up the variation (and clashes) of values and beliefs among respondent groups can be examined from a "conflict resolution" frame of reference. A first stage in conflict resolution calls for identification and explication of points of agreement and disagreement among affected parties. Also, the weights or values which parties attach to given issues are identified and made known to all participants. Armed with this information, it is assumed that the affected parties will be in a better position to work toward some level of consensus or compromise.

Decision making on an issue such as nuclear waste disposal may or may not be amenable to widely agreed-upon choices. The points of polarization among the population may be so great as to preclude a choice satisfactory to all concerned parties. The orientation of the authors, nevertheless, is that while more information of the type presented in this report may not simplify the decision maker's tasks, better decisions and more acceptable decisions are likely when it is available.

What is the methodological relevance of this research? What work remains to be done?

The purpose of this research was to assess public values and attitudes about a complex technological topic. The task was difficult because 1) the public's general level of awareness of the problem was relatively low and 2) there was no history of pretested methodologies for obtaining this type of information. The approach taken was to communicate a small body of information to respondents about the topic and then to elicit their judgments and attitudes about the topic via a survey questionnaire. Most respondents in the study voiced approval of the approach and the rationale used, although many pointed out areas where refinements in the approach were needed. Public input on other complex issues, if it is to be obtained early in the planning process, may require a variation of the methodology developed for this study. Providing respondents with scenarios which depict anticipated future alternatives and then obtaining their assessments of these alternatives may prove to be useful mechanisms for getting public input to crucial decisions. The research reported here constitutes an initial, exploratory effort using such an approach. Many of the results presented also suggest needed improvements and refinements in the method.

Another objective of this research was to test the utility and applicability of several measurement methods. For example, three measurement techniques for assessing relative importance were incorporated in the questionnaire. All three methods tended

to produce fairly comparable and consistent results, and the answer to 'which method is best?' may depend on the researcher's objectives. Many respondents expressed irritation with the policy capturing method in Part III of the questionnaire. This method may be a simulation, however, of the actual demands required in multi-attributed decision making, e.g., situations requiring several factors to be weighed and assessed simultaneously in deciding among alternatives. In other instances where the overriding question is one of relative priorities, a method similar to the ratio estimation procedure may be most useful.

This study focused on a set of value dimensions (i.e., Short-Term and Long-Term Safety, Cost, and Accident Detection and Recovery) which could be used to characterize all nuclear waste disposal options. Relatively little emphasis was placed upon the disposal options in the film, and the questionnaire did not ask respondents to judge the acceptability of disposal options. Respondents commented about this point and many expressed an interest in the actual disposal methods being considered. A useful area for future research might be directed toward furnishing public groups with information about specific methods, followed by an assessment of their attitudes and opinions of various methods. As the technical assessments and evaluations of disposal methods begin to narrow the range of options, public input regarding their acceptability could serve as vital information for decision makers.

Finally, the relevance for decision-making purposes of the information in this report should be assessed in a systematic fashion. Current follow-up activities call for the feedback of survey results to key groups such as planners, decision makers, and critics. The results from this feedback process will be vitally important in suggesting new directions for subsequent research on public values.

TASK 3. SPECIAL EVALUATIONS

No new special studies were started during FY 1976. As a result of restructuring of program activities, reports under this heading will be discontinued.

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2.0 DECONTAMINATION AND DENSIFICATION OF CHOP-LEACH CLADDING RESIDUES

R. L. Dillon and R. S. Kemper

Fuel element cladding residues from the chop-leach head end process constitute a low-density, high-surface area metallic waste with a substantial actinide element contamination. The objective of this study is to determine the feasibility of and to develop methods for removal of long half-life transuranics from the cladding residues and for consolidation of the zirconium and stainless steel residues. The study involves:

1. *Selection and laboratory testing of methods for transuranic removal by*
 - a) *aqueous reagents*
 - b) *molten salt reagents*
 - c) *fluxing during melting.*
2. *Consolidation of the fuel element hulls by melting and casting Zircaloy, stainless steel, Inconel or alloys of the three.*
3. *Testing and evaluating the consolidated fuel hulls for long-term waste management.*
4. *Absorption and fixation of tritium into the waste ingots.*
5. *Consolidation of transuranics removed.*

SUMMARY

Preparation of Sections 7 and 18 of the Technical Alternatives Document has occupied the time of the two lead engineers, almost to the exclusion of R&D activities. Limited work has been done on the mini-scale hot cell verification of the decontamination process--equipment assembly and checkout. Offsite effort has continued at the U.S. Bureau of Mines on Inductoslag melting and fabrication of a copper crucible for use at PNL.

A mini-scale melting apparatus has been constructed to determine the distribution of TRU elements between the calcium fluoride flux and the molten zirconium alloys. The high-vacuum glass/quartz gas handling system is operational, and the induction melting operation is being improved.

The demonstration during December that heavily oxidized sections of Zircaloy fuel tubing can be successfully melted by the Inductoslag process is an important development in waste volume reduction. Surface decontamination of fuel hull wastes will continue to be an essential step in minimizing the flow of material to engineered storage. The recent melting studies have shown, however, that oxide removal is not a necessary preliminary to successful melting.

TECHNICAL PROGRESS

Decontamination Process - B. Griggs

The distribution of TRU elements in Zircaloy fuel cladding has been previously described^(1,2) as being about 99.9 % on the surface of the metal. The remaining 0.1% of TRU

elements is produced from uniformly distributed uranium impurities in the original Zircaloy, allowed by ASTM up to 3.5 ppm. The removal of this 0.1% of the TRU contamination will usually be necessary to assure reduction of the TRU elements below 10 nCi/g even if the surface decontamination were complete. The removal of these trace levels (<10 ppb) of TRU elements will require extraction from the molten metal during densification. Further removal can be by evaporation, or by absorption on surfaces. The exploration of these extraction processes will be conducted in a vacuum melting apparatus which has been constructed in a hood in order to use plutonium, americium, uranium, and curium spiked samples of Zircaloy. The apparatus diagrammed in Figure 2 will be used

for melting and gas collection. Alpha counting, gamma energy analyses and mass spectrology will be used for stream analyses to determine the distribution of TRU elements in the molten system. The evolution and re-absorption of tritium will also be examined when melting irradiated hulls. The melting of stainless steel-Inconel-Zircaloy ingots will be under a magnesium-calcium fluoride slag to achieve a lower melting point flux consistent with the alloy melting range (950 to 1100°C).

Tests of the induction heater show problems in attaining the 1850°C melting point of Zircaloy. Improved coil design, smaller size, and better thermal shielding are being tested prior to increasing the power input to the system.

FIGURE 2. Induction Melting System

DECONTAMINATION FACILITY -

R. G. Moles

Procurement of PWR and BWR Fuel

Components - PWR fuel bundle end fitting and spacers have been located at Aerojet Nuclear Company in one of the fuel examination cells. A drawing of this bundle has been received. We are currently deciding how much of the end fitting will be needed for decontamination and densification studies.

The BWR fuel rod previously located was determined to have a reactor core exposure somewhat lower than desired. A fuel rod having at least 18,000 MWd/T should be used in the decontamination studies to assure that thick oxides can be satisfactorily removed. The rod previously located had an exposure of around 16,000 MWd/T. Another rod has been located in storage that has around 25,000 MWd/T. Verbal communications with General Electric have indicated that this rod is fairly accessible, and a letter is being prepared to PNL confirming reactor exposure and cost of shipping preparations.

1706KEL Mini-Scale Decontamination

Facility - The engineering analysis by PNL Facilities Engineering for the emergency hot cell exhaust ventilation has been completed. The analysis shows that the 110 volt DC power supply may be slightly cheaper than a small automatic starting, gasoline motor driven generator delivering 110 V AC. The DC supply has been chosen for installation. The mechanical design for ducting modification and blower installation has started and should be completed by February 13, 1976.

Mini-Scale Decontamination Facility -

Several tests have been conducted on the heating of the chamber in which the oxidized Zircaloy contacts Ar-HF gas mixtures. Vessel heating has been stabilized, and good temperature control has been achieved. However, the trap for scavenging O_2 from the Ar stream heating continues to give temperature control problems. When process temperatures of $575^{\circ}C$ have been obtained with the Ar gas stream, inside wall temperatures of the trap are about $75^{\circ}C$ higher ($650^{\circ}C$). With voltage and current remaining constant, the gas stream temperature starts to rise. Process temperatures have risen to as high as $700^{\circ}C$. The O_2 trap was disassembled, and the Zr chips and thermocouples were examined. The Zr chip material originally used was fairly fine machine turnings. These have now been replaced with coarser turnings and chips. Two new Inconel clad thermocouples are being fabricated and are scheduled for delivery by January 20, 1976.

DENSIFICATION PROCESS - R. G. Nelson

Work to date has been concentrated on the melting of cleaned cladding hulls for direct storage or, in the case of pure Zircaloy, for reuse. It is conceivable that direct melting of uncleaned hulls may be a desirable approach for volume reduction and storage.

Highly oxidized Zircaloy-4 was prepared and melted by the Inductoslag process to determine the ability of the process to densify uncleaned hulls. Four autoclave runs

of about 10 kg each were made with 90-cm-long pieces of reject Zircaloy-4 clad tubing. The autoclave conditions were 500 to 525°C, 40 to 48 hr and 200 psi. The four batches produced showed oxygen increases of 2900, 3500, 3700 and 7700 ppm. Subsequent chopping into one-inch-long pieces revealed the somewhat brittle nature of the material. Each batch was melted separately, one atop the other, to produce a single 10-cm diameter ingot about 80 cm long. Figure 3 shows this ingot before and after light scraping to remove the calcium fluoride slag. Melting characteristics appeared nearly identical to those of clean hulls. The high (7700 ppm) oxygen batch showed more splattering and smoke than the others; however, melting was not inhibited. Melting at 85 kW resulted in a melt rate of about 0.34 kg/min. The overall appearance of the ingot is nearly

identical to one melted with clean hulls, with the surface finish of the high-oxygen material being less shiny and more matte-like. cursory hardness testing with a file indicates the high-oxygen material to be much harder than the low-oxygen material.

The results of this experiment indicate that the Inductoslag process can melt without difficulty highly oxidized Zircaloy cladding.

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FIGURE 3. Inductoslag Ingot Heat Number 529 Melted From Highly Oxidized Zircaloy-4 Cladding Hulls

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3.0 PYROLYSIS-INCINERATION OF COMBUSTIBLE ALPHA WASTES

L. K. Mudge and P. C. Walkup - Project Managers

The principal program objective is to develop a pyrolysis-incineration process for safe and economical reduction of the volume of alpha wastes from 10- to 50-fold as a precursor to ultimate disposal. The technical and economic merits of this process will be defined to allow its comparison to other processes being considered for volume reduction of alpha wastes. The program objective will be pursued via 1) a phased program of bench-scale research leading to supporting pilot plant development activities and 2) systematic and continuing analytical studies to assess the technical and economic merits of the process. Bench-scale and pilot plant activities will be conducted first with noncontaminated simulated wastes and later with actual wastes from Hanford operations.

SUMMARY

The pilot plant equipment was operated continuously over a 3-day period in November. Simulated alpha waste was successfully processed in the pyrolysis-incinerator over a 48-hr period at an average rate of 7 kg/hr with an air flow of 198 liters/min.

Gases produced during processing of the simulated waste were flared by use of a makeshift burner fashioned from pieces of pipe. No auxiliary fuel was required to maintain combustion of the gases, which flared with a clean flame.

Equipment performance was quite satisfactory. Plugging of lines remote from the pyrolysis-incinerator by accumulation of tars did occur. This problem could be alleviated by installation of a gas burner closely coupled to the pyrolysis-incinerator.

TECHNICAL PROGRESS

Pilot Plant Studies - L. K. Mudge,
G. E. Stegen, D. G. Ham,
G. F. Schiefelbein

The pyrolysis-incineration pilot plant was operated continuously over

a 3-day period for the volume reduction of simulated alpha waste. The unit was shut down several times during the run for times up to 75 min as the result of an electrical outage and to clean off gas lines and repair the seal on the distributor shaft. After each shutdown, processing was resumed by simply starting air flow to the pyrolysis-incinerator.

Simulated alpha waste composed of 10% wood, 15% paper, 15% rags, 20% PVC, 15% polyethylene, 20% neoprene rubber and 5% latex rubber was used in the run. The wood, paper, rags and rubber were shredded as described in earlier work⁽¹⁾ while the polyethylene and PVC were pieces about 3 mm in size.

Startup and operation of the pilot plant was similar to procedures used in earlier runs.⁽¹⁾ Unlike earlier runs, the bed depth in the pyrolysis-incinerator was not allowed to pass a point 20 cm below the top of the refractory lining. Waste was mixed and added to the unit in 5-kg batches at time intervals as required to maintain the desired bed depth.

A total of 331 kg of simulated waste was processed over a 48-hr period for an average processing rate of 7 kg/hr. At times during the run, rates of 15 kg/hr were required to maintain the desired bed level in the pyrolysis-incinerator. Air flow was maintained at a constant rate of 198 liters/min at standard conditions.

Operation of the pilot plant was quite satisfactory during the run. Further reinforcement of the distributor is necessary because lateral movement during the rotation and vertical traverse of the distributor is still evident. This lateral movement caused the seal to be torn from its seat, requiring that the unit be shut down three times for seal repair during the run. It is possible to redesign the distributor to eliminate all lateral movement. Offgas lines were cleared by steam injection during these shutdowns. It was necessary to

shut down once to remove tar from the elbow immediately off the cover on the pyrolysis-incinerator. Installation of a gas burner coupled in close proximity to the pyrolysis incinerator would virtually eliminate problems with line plugging by tars.

Tars collected during the run totaled 46 kg, or 14% of the weight of the feed processed. The viscosity of the tars comprising most of the total weight was 30 cp at 75°C. Ash and residue remaining in the reactor weighed 21 kg. About 7 kg of ash was obtained for fixation studies.

Gases produced during the run were burned cleanly in a small burner, fashioned from pipe fittings for a 3 1/2-hr period during the run. This test confirmed the concept of a small, close-coupled burner for direct combustion of tars and gases. Gas compositions obtained during the run are shown in Table 2. Examination of

TABLE 2. Gas Compositions for Run 6

Time	% By Volume								
	H ₂	CO ₂	C ₂ H ₄	C ₂ H ₆	Ar-O ₂	N ₂	CH ₄	CO	C ₃ H ₈
11-3-75									
1315	2.9	12.3	-	-	0.6	66.9	-	17.3	-
Start waste feed									
1357	5.1	13.1	0.6	0.2	0.5	62.9	1.6	15.7	0.1
1700	6.0	19.0	0.6	0.4	0.5	67.5	1.9	3.5	0.5
2000	7.3	16.6	1.4	0.2	0.5	67.1	1.9	5.0	-
2325	6.2	19.0	1.0	0.2	0.5	68.0	1.3	3.3	0.4
11-4-75									
0130	8.5	13.6	1.2	0.3	0.5	61.7	2.1	11.8	0.2
0350	7.5	9.5	0.4	0.2	0.5	59.3	1.5	21.0	0.1
0600	8.2	13.1	0.2	0.3	0.5	60.1	2.0	15.6	0.1
1300	6.5	6.5	0.2	0.2	0.5	57.6	1.0	27.6	-
2300	7.9	5.6	0.4	0.2	0.5	35.9	1.5	27.8	0.2
11-5-75									
0300	6.5	13.8	0.2	0.2	0.5	63.9	1.2	13.7	0.1
1135	8.7	8.0	0.9	0.3	0.4	54.6	1.9	24.8	0.3

the CO concentration indicates that, after the start of the simulated waste feed, considerable time is required to reestablish the char zone. This is shown by a decreased CO concentration followed by an increase in CO concentration. A low CO concentration indicates channeling through the bed of refuse in the pyrolysis-incinerator.

The maximum temperature in the pyrolysis-incinerator recorded during the run was 960°C. Gases leaving the unit never exceeded 100°C.

Ash collected during the run had fused into small nuggets 1 to 3 cm in

size. The ash contained 0.7 wt% unburned carbon. Analysis of this ash is currently being conducted.

Ash Fixation Studies -

J. E. Mendel and H. Hollis

No ash samples were available during the quarter for fixation studies. Ash from the extended run will be studied for conversion to a glass.

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4.0 TRITIUM SEPARATION AND FIXATION

L. L. Burger - Project Manager

The goals of the program are to investigate isotopic separation as a means of concentrating tritiated aqueous wastes, to consider alternative procedures such as increased recycle of water, and to examine methods of fixing tritiated water for permanent storage.

SUMMARY

Thermal stabilities of polymers for tritium fixation were studied using thermogravimetric analysis (TGA). The development of a process for the fixation of THO in a polymer within minutes at room temperature, including optimization of the polymer formulation for molding and leach testing of tritiated samples, was completed. Development of a procedure for TH fixation in styrene was delayed by difficulties with air oxidation of the homogeneous catalyst system.

Several hydride samples have been prepared starting with zircaloy metal. High impurity levels, notably appreciable amounts of oxygen, nitrogen, and carbon, and slow reaction rates have been consistent problems. Further changes in techniques have been made to improve the reaction.

TECHNICAL PROGRESS

Organic Polymers - J. A. Franz

Most organic polymers decompose at $<300^{\circ}\text{C}$.^(1,2) Some polymers, such as polyamides, are stable up to about 550°C . Since thermal behavior of the fixation media is important to transportation and terminal storage, thermogravimetric analyses (TGA) were

carried out on polymers of interest.

A resorcinol-acetaldehyde-formaldehyde polymer described earlier⁽³⁾ undergoes heat aging and weight loss at $>130^{\circ}\text{C}$ and loses 50% of weight by 390°C , with a heating rate of $25^{\circ}/\text{min}$. This closely parallels the behavior of phenol-formaldehyde resins.^(1,2) Aromatic polyureylene undergoes decomposition at $>200^{\circ}$. All of the polymers underwent nearly complete vaporization ($\geq 95\%$) on heating above 500°C .

Table 3 provides representative data.

The development of a process for the direct, quantitative incorporation of water at room temperature in an organic polymer was completed this month. Full details of the procedure will be described later in a forthcoming supplement to BNWL-B-430.⁽³⁾ Briefly, tritiated water and two other organic components are combined in an optimized ratio in acetone solvent. Precipitation of the polymer begins immediately and is complete within one hour. Removal of solvent under vacuum leaves a white powder, in which water is quantitatively fixed. The polymer is molded in a cylindrical mold at 200°C under 20,000 psi pressure for 15 min to give a semitransparent, amber-

TABLE 3. Thermogravimetric Analyses of Polymers

Polymer	Atmo- sphere	Onset of Weight Loss, T	50% Weight Loss, T	Rate of Heating, °C/min
Polystyrene ^(a)	Ar	280-300°C	430°C	25
	Air	280-300	390	25
Poly(hexamethylene)	Ar	270-289	380	25
	Air	230-240	380	25
Poly(2,4-tolylidene ureylene)	Ar	180-200	300	25
	Air	180-200	335	25
Bakelite ^(b) analogue ^(b)	Ar	>130 ^(c)	390	25
	Air	>130 ^(c)		

a. Model for polystyrene-polyvinylcyclohexane polymer.

b. Resorcinol-acetaldehyde-formaldehyde polymer.

c. Heat aging begins at >130°. Rapid decomposition begins at ~250°.

colored solid of density 1.15 g/cm³. The material is not friable, but is sufficiently brittle to fracture under conditions of severe shock. Uniform, thorough heating during molding is necessary to avoid strain-cracking in the product. Tritiated samples of the powdered and molded polymer were prepared. The tritiated molded polymer lost <4% of total activity on leaching for 7 days. An extended leach test is under way.

The procedure is an extremely simple, rapid, and quantitative method for the fixation of tritium in an organic polymer. Unit operations are simple, and low capital costs are anticipated for the process. The monomer starting materials are manufactured for use in the polymer industry on a large scale, and the procedure is a simple modification of well established synthetic methods for polymers. The material properties of the product are excellent. The overall

cost, about \$5 per pound of water, is slightly higher than for other polymers.

A procedure for the fixation of TH in styrene via homogeneous catalysis was delayed by difficulties with air oxidation of the catalyst. No further work on this system will be carried out.

Metal Hydride Fixation - R. D. Scheele

The general approach to be used in this study was outlined in the previous progress report BNWL-1952.⁽⁴⁾

The preparation of a metal hydride is influenced by a number of factors, "(1) impurities in the metal, (2) surface condition of the metal (oxide, nitride films, etc.), (3) impurities in hydrogen, (4) form of the metal to be hydrided, and (5) pre- and post-heat treatment, as well as by pressure-temperature relations."⁽⁵⁾

Zirconium is a metal which is easily hydrided. Zircaloy 2 and 4

are formed by alloying zirconium with small amounts of tin, nickel, iron, and some other minor constituents. The work of M. J. Trzeciak, D. F. Dilthey, and M. W. Mallet⁽⁶⁾ and others has indicated some of the effects of alloying zirconium with various metals on the formation of their hydrides. In some cases, e.g., nickel in low concentration, the attainable H/M ratio was increased; in other cases, e.g., tin in both high and low concentrations decreased this ratio. The stability with respect to temperature of a hydride formed from a metal alloy, 2.3a% transition element and 97.7a% Zr, did not differ greatly from that of pure zirconium. Zircaloy should be a suitable material for tritium fixation.

Six hydriding experiments were carried out using 0.5-in. pieces of 0.25-in. diam Zircaloy 2 rod. One hydride was made from a 0.5-in. piece of 0.5-in. diam Zircaloy 2 tubing. Table 4 gives the composition of each of the samples prepared.

The hydrides were made using 99.5% pure hydrogen, Matheson "Hydrogen, Prepurified" grade. It was assumed this was a sufficient level of purity of H₂ and required no further purification. The metal was obtained from Teledyne Wah Chang and was used with no surface preparation. Auger analysis of a prepared hydride indicated high surface content of oxygen, nitrogen, and carbon, suggesting that surface preparation may be required to remove the oxide and nitride layers by fine sanding and washing with reagent grade acetone for removal of grease. It is, of course, necessary to provide a completely leak-tight hydriding apparatus. Sample analyses show high levels of oxygen and nitrogen present. This condition could arise from a leak in the hydrogen input line or in the hydriding apparatus. At this time, no leak in the latter system has been detected using a mass spectrometer helium leak detector.

TABLE 4. Composition of Prepared Hydrides

Sample No.	Initial Mass, g	Mass Gain, g	Composition Ratios by Analysis				
			H:Zr	O:Zr	N:Zr-3 x 10 ⁻³	C:Zr-4 x 10 ⁻⁴	H:Zr (a)
1	6.4719	0.0225	0.1019:1	0.01346:1	5.545:1	5.069:1	0.160:1
2	3.1711	0.0289	0.1928:1	0.02345:1	3.845:1	7.213:1	1.45:1
3	3.0950	0.0433	0.6192:1	0.02199:1	2.253:1	5.609:1	--
4	3.2204	0.0613	0.02691:1	0.09047:1	3.889:1	9.558:1	--
5	3.0003	0.1031	0.2023:1	0.02124:1	5.457:1	8.718:1	--
6	3.2556	0.0326	0.2128:1	9.944E-03:1	4.079:1	4.274:1	--
7	6.3818	0.0539	(b)	(b)	(b)	(b)	0.60:1

a. Determined by difference in the amount of hydrogen added and hydrogen remaining after hydriding.

b. Has not been analyzed at this time.

Several techniques were employed in an attempt to improve the hydriding rate and the total uptake of hydrogen. The method used in recent Brookhaven⁽⁷⁾ work was tried initially. This employs isothermal adsorption at subatmospheric hydrogen pressures. Blackledge⁽⁵⁾ describes a method involving addition of hydrogen at the outgassing temperature then reduction of the temperature as the adsorption begins. He also suggests slow cooling of the sample under H_2 and thermal cycling. J. G. McGuire of HEDL also indicated the importance of thermal cycling. The literature has also suggested activation by adding and removing hydrogen at 400°C or above but this has not always proved useful. It is evident from our experiments and from other reported work that hydriding is easily inhibited, primarily by impurities. Although rates can be improved by conditions that promote surface cracking, systems reasonably free from impurities will probably be required for acceptable hydriding of zirconium.

Analysis of the samples for hydrogen, oxygen, nitrogen, and carbon content was performed by HEDL Analytical Laboratory. Difficulty in obtaining a homogeneous sample of the hydride should be noted. At elevated temperatures oxygen, nitrogen, and hydrogen will be driven to the interior of the sample over a period of time.⁽⁵⁾

Thus, if insufficient time is allowed for equal distribution of these elements throughout the hydride, a representative sample cannot be achieved.

Samples 1 through 6 were grey to black indicating high nitrogen and

oxygen content. Sample 7 had a golden hue indicating a slight amount of N_2 absorption.⁽⁵⁾

Pyrophoricity has been of concern in consideration of metal fixation of tritium. The unpredictable pyrophoric nature of zirconium has been known for many years. However, little is known about the mechanism.^(8,9)

Contributing factors to pyrophoricity are the form of the metal, (chips and fines show high pyrophoricity due to their high surface area), and the presence of water vapor and other gases including hydrogen.^(8,9) Blackledge⁽⁵⁾ states that titanium, zirconium, and yttrium hydrides are less prone to spontaneous ignition in air than are their parent metals. The evident cause of this is the slow formation of an oxide layer on the outer surface. Thermodynamically the hydride is slightly less stable towards oxidation than is the metal; further studies need to be made.

Future work will attempt to improve the kinetics by reducing the impurities and will involve further use of Zircaloy rods and zirconium rods and chip forms.

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5.0 DISPOSITION (D&D) OF RETIRED CONTAMINATED FACILITIES AT HANFORD

K. M. Harmon - Project Manager

The program has the following objectives for FY 1975-1976: to establish methods, costs and priorities for the disposition of retired (excessed) Hanford facilities; to formulate plans and specifications for a project to accomplish permanent disposition of a specific Hanford facility; and to identify required facility disposition R&D.

SUMMARY

Radiological criteria governing the release and use of a reactor retention basin system and the Redox Plant plutonium concentration (233-S) building were developed, for four different use restriction categories. A computer-based management information system for the contaminated facilities at Hanford was developed and input provided for the liquid disposal sites, burial grounds, and reactor retention basin systems. This system, to be used for setting D&D priorities for retired contaminated Hanford facilities, was used to derive preliminary priorities among the various retention basin systems. Planning for the 233-S D&D demonstration continued. The demonstration trench to be used in a test of a proposed technique for stabilizing contaminated ground sites was installed as planned.

TECHNICAL PROGRESS

TASK 1. Establish Methods, Costs and Priorities for D&D at Hanford -
K. M. Harmon
Radiological Disposition Criteria -
D. A. Waite

A technique⁽¹⁾ for deriving radiological criteria governing the release and use of decommissioned nuclear facilities was applied to two retired contaminated Hanford facilities, a reactor retention basin system (107-DR) and the Redox Plant plutonium concentration building (233-S). The results of the study, shown in Tables 5 and 6, provide maximum radionuclide concentration levels for these four use restriction categories:

- Restricted use - the facility or site is to be limited to nuclear use only, with access allowed only under conditions normal to nuclear sites.
- Controlled use with surveillance - use of the site is to be limited in a manner which depends on the nature of the facility; continuing surveillance is required.
- Controlled use without surveillance - use of the site is to be limited to appropriate activities, by laws or regulations.
- Unrestricted use - the site may be unconditionally released for access and use by the public.

TABLE 5. Disposition Criteria - 107 DR

		Surface Contamination*	Activity Mass Concentration*
Conditional	Unrestricted	$2.1 \alpha + \beta \lambda = 460 \text{ dpm/100 cm}^2 \text{ fixed}$ $2.1 \alpha + \beta \lambda = 460 \text{ dpm/100 cm}^2 \text{ smearable}$	$2.1 \alpha + \beta \lambda = 21 \text{ pCi/g}$
	Nonsurveillance	$2.1 \alpha + \beta \lambda = 4.6 \times 10^3 \text{ dpm/100 cm}^2 \text{ fixed}$ $2.1 \alpha + \beta \lambda = 460 \text{ dpm/100 cm}^2 \text{ smearable}$	$2.1 \alpha + \beta \lambda = 210 \text{ pCi/g}$
	Surveillance	$2.1 \alpha + \beta \lambda = 4.6 \times 10^4 \text{ dpm/100 cm}^2 \text{ fixed}$ $2.1 \alpha + \beta \lambda = 460 \text{ dpm/100 cm}^2 \text{ smearable}$	$2.1 \alpha + \beta \lambda = 2.1 \text{ nCi/g}$
	Restricted	$2.1 \alpha + \beta \lambda = > 4.6 \times 10^4 \text{ dpm/100 cm}^2 \text{ fixed}$ $2.1 \alpha + \beta \lambda = 460 \text{ dpm/100 cm}^2 \text{ smearable}$	$2.1 \alpha + \beta \lambda = > 2.1 \text{ nCi/g}$

* α and $\beta\gamma$ denote alpha and beta-gamma activities in the same units as appear on the right side of the equation: dpm/100 cm², pCi/g or nCi/g.

TABLE 6. Disposition Criteria - 233-S

		Surface Contamination	Activity Mass Concentration
Conditional Use	Unrestricted	1.8×10^3 dpm/100 cm ² fixed 1.8×10^3 dpm/smearable	80 pCi/g
	Nonsurveillance	1.8×10^4 dpm/100 cm ² fixed 1.8×10^3 dpm/smearable	800 pCi/g
	Surveillance	1.8×10^5 dpm/100 cm ² fixed 1.8×10^3 dpm/smearable	8.0 nCi/g
	Restricted	$> 1.8 \times 10^5$ dpm/100 cm ² fixed 1.8×10^3 dpm/smearable	8.0 nCi/g

Computer-Based Information System -

J. C. King and J. W. Litchfield

A computer-based management information system has been developed for the contaminated facilities at Hanford, based on the DML language of the INFONET system. Available information has been entered for the contaminated liquid disposal sites (cribs, trenches, ponds, etc.), the contaminated solids storage and disposal sites (burial grounds), and the reactor retention basin systems.

The main use of the system in the D&D program will be in establishing priorities. For example, early D&D attention will need to be given to cribs and burial grounds which, because of their physical characteristics, are susceptible to incursion by burrowing animals or tumbleweeds. Tables 7 and 8 list the liquid disposal sites selected by the computer for this characteristic.

TABLE 7. Liquid Disposal Sites with Potential for Animal Dispersion of Radionuclides

CRITERIA: DEPTH LESS THAN TEN FEET
HIGH SALT CONTENT
FISSION PRODUCT CONCENTRATION > .04 CI/SQ M
RANKED BY FISSION PRODUCT CONCENTRATION, DEPTH

FACILITY DESIGNATOR	FACILITY NAME	DEPTH FT	F.P. CONCN CI/SQ M	CONSTRUCTION	AREA SQ FT	ACID	BETATOTAL CI	TRU CONCN GM/SQ M
216-T-25	III-B CRIB	10	65.0000	EXCV FILL	1800	N-B	11,000.0000	.0100
216-B-37	III-B TRENCH	10	16.0000	EXCV FILL	2520	N-B	3,800.0000	.0100
216-R-38	III-B TRENCH	10	12.0000	EXCV FILL	2520	N-B	2,900.0000	.0100
216-B-30	IIIB CRIB	6	11.0000	WOOD	5000	N-B	5,200.0000	< .0100
216-T-22	III-B CRIB	10	10.0000	EXCV FILL	2400	N-B	2,300.0000	< .0100
216-T-18	III-B CRIB	10	0.3000	CONC	100	N-B	77.6000	190.0000
216-T-24	III-B CRIB	10	0.0200	EXCV FILL	2400	N-B	1,800.0000	.0100
216-T-23	III-B CRIB	10	7.5000	EXCV FILL	2400	N-B	1,700.0000	< .0100
216-B-36	III-B TRENCH	10	6.8000	EXCV FILL	2500	N-B	1,600.0000	.0100
216-B-20	IIIB CRIB	6	6.5000	WOOD	5000	N-B	3,000.0000	< .0100
216-B-42	II-B TRENCH	10	6.4000	EXCV FILL	2520	N-B	1,500.0000	.0400
216-T-15	III-B CRIB	10	5.8000	N/A	2400	N-B	1,300.0000	< .0100
216-B-26	IIIB CRIB	6	5.0000	WOOD	5000	N-B	2,700.0000	.0100
216-B-41	III-B TRENCH	10	5.1000	EXCV FILL	2520	N-B	1,200.0000	< .0100
216-B-35	IIIB TRENCH	10	3.5000	EXCV FILL	2520	N-B	810.0000	.0100
216-B-40	III-B TRENCH	10	3.3000	EXCV FILL	2520	N-B	780.0000	< .0100
216-B-21	IIIB CRIB	6	3.2000	WOOD	5000	N-B	1,500.0000	.0200
216-T-16	III-B CRIB	10	2.9000	EXCV FILL	2400	N-B	650.0000	< .0100
216-T-14	III-B CRIB	10	2.8000	GRAV	2200	N-B	580.0000	< .0100
216-B-39	III-B TRENCH	10	2.5000	EXCV FILL	2520	N-B	500.0000	.0100
216-T-21	III-B CRIB	10	2.3000	EXCV FILL	2400	N-B	510.0000	< .0100
216-T-17	III-B CRIB	10	2.1000	N/A	2750	N-B	460.0000	< .0100
216-B-22	IIIB CRIB	6	1.4000	WOOD	5000	N-B	630.0000	.0100
216-B-32	IIIB CRIB	6	1.1000	WOOD	5000	N-B	510.0000	.0100
216-B-24	IIIB CRIB	6	.9200	WOOD	5000	N-B	430.0000	.0200
216-B-33	IIIB CRIB	6	.9000	WOOD	5000	N-B	420.0000	.0300
216-B-52	II-B CRIB	8	.8700	EXCV FILL	5800	N-B	470.0000	.0100
216-B-29	IIIB CRIB	6	.7700	WOOD	5000	N-B	360.0000	< .0100
216-B-25	IIIB CRIB	6	.7700	WOOD	5000	N-B	360.0000	< .0100
216-B-31	IIIB CRIB	6	.7100	WOOD	5000	N-B	330.0000	.0100
216-B-28	IIIB CRIB	6	.4100	WOOD	5000	N-B	190.0000	.0100
216-B-34	IIIB CRIB	6	.1700	WOOD	5000	N-B	79.0000	.0100

The cribs and trenches in which high-salt waste solutions were discharged at depths of 10 feet or less and in which the total fission products (corrected for decay to 12/73) discharged to the ground averaged $>0.04 \text{ Ci/m}^2$ are presumed to have high potential for the spread of contamination by burrowing animals. Table 7 ranks these sites in order of decreasing fission product concentrations.

The sites and ranking in Table 8, which identifies the liquid waste disposal sites with a potential for the growth of radioactive tumbleweeds, were selected on the basis of a model based on the following assumptions:

- Tumbleweed contamination is entirely attributable to uptake of ^{90}Sr .
- Tumbleweeds having a ^{90}Sr concentration equal to or exceeding $1 \times 10^{-9} \text{ Ci/g}$ dry weight are considered as contaminated.
- Transfer of ^{90}Sr from soil to weed is assumed to be on 1:1 basis (weight of soil to dry weight of weed), weighted by tumbleweed root distribution through the soil profile. Ninety percent of tumbleweed roots were assumed to lie in the first 2 ft of the soil profile; the remaining 10% of the roots were assumed to be evenly distributed through the next 18 ft of the soil profile.
- Concentration of ^{90}Sr in the soil profile was assumed to originate from even distribution of the 1973 ^{90}Sr inventory through a

volume of soil defined by the surface area of the bottom of the facility and a depth of 20 ft below the bottom of the facility.

D&D Priorities - J. C. King, J. W. Litchfield and B. W. Mercer

Criteria establishing facility priorities have been further refined and reduced to four:

- Offsite potential radiological hazard
- Onsite potential radiological and physical hazard
- Cost of maintenance and surveillance
- Compatibility with future use of site environs.

A set of programs has been written to facilitate the setting of priorities.

- SCORE is an interactive program facilitating rapid entry of facility performance scores (on criteria selected for prioritization) into the D&D files.
- WEIGHT is an interactive program which solicits raw criteria weights of two or more individuals, as derived from pairwise comparison of criteria, and transforms these weights into normalized weights for the group.
- PRIOR 1 computes priority indices for a set of facilities using performance scores and prioritization criteria weights. Indices are stored separately from the D&D files. PRIOR 2 is an associated program which reports facility number, name, status, performance scores and priority indices for individual facilities, in order of priority.

TABLE 8. Liquid Waste Disposal Sites with Potential Problem of Radioactive Weeds

CRITERIA: CONCENTRATION IN WEED > 1 E-9 CI/ GM, WHERE
 45% OF ROOTS IN EACH OF TOP TWO FEET OF SOIL
 .5% OF ROOTS IN EACH OF NEXT 18 FEET OF SOIL
 FACILITY DEPTH LESS THAN 20 FEET
 ***, MEANING UNKNOWN DEPTH, TREATED AS 0
 RANKED BY AREA, PROBLEM INDEX

FACILITY DESIGNATOR	FACILITY NAME	DEPTH FT	SR 90 CI	CONSTRUCTION	AREA SQ FT	ACID	BETATOTAL CI	F.P. CONCEN CI/SQ M	TRU CONCEN GM/SQ M	PROBLEM INDEX
216-B-2-2E	VI DITCH	6	650.0000	EXCV	20000	UNK	1,300.0000	U	U	.00001
216-B-2-2W	VI DITCH	6	220.0000	EXCV	14000	UNK	450.0000	U	U	.00001
216-T-34	II-C CRIB	16	270.0000	GRAV	6000	N-B	1,300.0000	2.3000	.1700	.00001
216-A-36B	II-C CRIB	***	500.0000	GRAV	5500	N-B	5,700.0000	11.0000	.3500	.00019
216-B-26	IIIB CRIB	6	720.0000	WOOD	5000	N-B	2,700.0000	5.8000	.0100	.00012
216-B-20	IIIB CRIB	6	510.0000	WOOD	5000	N-B	3,000.0000	8.3000	.0100	.00000
216-B-21	IIIB CRIB	6	480.0000	WOOD	5000	N-B	1,500.0000	3.2000	.0200	.00008
216-B-30	IIIB CRIB	6	400.0000	WOOD	5000	N-B	5,200.0000	11.0000	.0100	.00007
216-B-27	IIIB CRIB	6	400.0000	WOOD	5000	N-B	870.0000	1.7000	.0100	.00003
216-B-22	IIIB CRIB	6	370.0000	WOOD	5000	N-B	630.0000	1.4000	.0100	.00004
216-B-32	IIIB CRIB	6	170.0000	WOOD	5000	N-B	510.0000	1.1000	.0100	.00003
216-B-31	IIIB CRIB	6	140.0000	WOOD	5000	N-B	330.0000	.7100	.0100	.00002
216-B-29	IIIB CRIB	6	130.0000	WOOD	5000	N-B	360.0000	.7700	.0100	.00002
216-B-25	IIIB CRIB	6	130.0000	WOOD	5000	N-B	360.0000	.7700	.0100	.00002
216-B-24	IIIB CRIB	6	120.0000	WOOD	5000	N-B	430.0000	.9200	.0200	.00002
216-B-23	IIIB CRIB	6	94.0000	WOOD	5000	UNK	350.0000	.7500	.0100	.00002
216-B-28	IIIB CRIB	6	75.0000	WOOD	5000	N-B	190.0000	.4100	.0100	.00001
216-B-42	II-B TRENCH	10	700.0000	EXCV	2520	N-B	1,500.0000	6.4000	.0400	.00017
216-B-40	II-B TRENCH	10	170.0000	EXCV	2520	N-B	780.0000	3.3000	.0100	.00004
216-B-35	II-B TRENCH	10	150.0000	EXCV	2520	N-B	810.0000	3.5000	.0100	.00004
216-B-36	II-B TRENCH	10	300.0000	EXCV	2500	N-B	1,600.0000	6.8000	.0100	.00007
216-A-27	II-C CRIB	***	37.0000	STEEL	2000	N-B	260.0000	1.4000	.5200	.00004
216-B-16	II-B CRIB	13	460.0000	STEEL CONC	2000	N-B	1,800.0000	12.0000	.0700	.00010
216-A-8	IIIC CRIB	14	77.0000	ROCK	1760	N-B	1,700.0000	1.1000	.0300	.00002
216-B-14	II-B CRIB	13	260.0000	STEEL CONC	1600	N-B	870.0000	5.9000	.1700	.00007
216-B-19	II-B CRIB	13	130.0000	STEEL CONC	1600	N-B	620.0000	4.2000	.0700	.00004
216-B-15	II-B CRIB	13	130.0000	STEEL CONC	1600	N-B	530.0000	3.7000	.0300	.00004
216-B-18	II-B CRIB	13	120.0000	STEEL CONC	1600	N-B	580.0000	3.9000	.0700	.00003
216-B-17	II-B CRIB	13	100.0000	STEEL CONC	1600	N-B	490.0000	3.3000	.0700	.00003
216-B-46	II-B CRIB	14	950.0000	STEEL CONC	900	N-B	2,200.0000	26.0000	.2400	.00041
216-B-43	II-B CRIB	14	870.0000	STEEL CONC	900	N-B	2,100.0000	25.0000	.0100	.00037
216-B-48	II-B CRIB	14	830.0000	STEEL CONC	900	N-B	2,100.0000	27.0000	.0600	.00036
216-B-47	II-B CRIB	14	370.0000	STEEL CONC	900	N-B	980.0000	17.0000	.0600	.00017
216-T-26	II-B CRIB	15	425.0000	CONC	900	UNK	1,090.0000	13.0000	.7000	.00016
216-T-28	II-C CRIB	15	160.0000	CONC	900	N-B	1,400.0000	17.0000	.8400	.00006
216-T-27	II-C CRIB	15	110.0000	CONC	900	N-B	400.0000	4.7000	.1600	.00004
216-C-3	III-A CRIB	10	12.0000	GRAV	500	ACD	25.0000	.5300	.0200	.00001
216-C-6	III-A CRIB	16	44.0000	GRAV	200	ACD	91.0000	4.9000	.0100	.00006
216-C-4	II-C CRIB	16	18.0000	GRAV	200	N-B	36.0000	1.9000	.0500	.00002
216-C-1	II-B CRIB	13	130.0000	CONC	184	N-B	270.0000	.1600	.6200	.00031
216-C-10	III-A CRIB	7	52.0000	RUCK	160	ACD	105.0000	7.1000	.0100	.00025
216-T-12	II-C CRIB	8	3.1000	EXCV	150	N-B	19.0000	1.3000	.0700	.00001
216-T-18	II-B CRIB	10	4.2300	CONC	100	N-B	77.6000	8.3000	.1900	.00003
216-C-5	II-B CRIB	16	6.4000	GRAV	100	UNK	13.0000	.6900	.0500	.00002
216-A-24	IIIC CRIB	15	28.0000	ROCK	28	N-B	814.0000	.3100	.0100	.00033

The manner in which the prioritization system is to be applied is illustrated by the following sequence of activities conducted for the retention basins:

Weights on the four criteria selected for prioritization were established by representatives of ERDA/RL and the major Hanford contractors using a DELPHI approach with two iterations. The group weights, normalized to a total of 100 points, were:

Onsite potential radiological and physical hazard	40.7
Offsite potential radiological hazard	24.6
Compatibility with future use of site environs	23.2
Cost of maintenance and surveillance	11.6

Performance of each facility with respect to each of the four criteria was estimated by BNW personnel, using maps, D&D information system reports, recent photos of the individual facilities, and a limited field inspection. Performance was estimated on a scale of 1 to 10, with 1 being "good" performance (low potential hazard, low cost and high compatibility) and 10 being "poor" performance (high potential hazard, high cost, poor compatibility). All estimates are relative to the performance of the 107-DR Retention Basin, which was assigned a value of 5 for each criterion. Resulting estimates of performance are listed in Table 9 under the respective criterion column heading.*

A priority index (Table 9) was computed for each facility using programs PRIOR 1 and PRIOR 2 and the following scheme:*

$$I_j = \sum_{i=1}^4 w_i P_{ij}$$

where I_j is an index of priority for facility j

w_i is the weight of criterion i

P_{ij} is the performance of facility j with respect to criterion i .

TASK 2. Formulate Plans and Specifications for a D&D Demonstration - K. M. Harmon and D. E. Blahnik
Detailed Planning - R. E. Smith (ARHCO)

Draft work procedures covering field activities required by Activity Specification I were prepared and reviewed by Atlantic Richfield Hanford Company staff. This initial program effort includes a complete radiation survey of the facility and a visual inspection to determine unusual hazards or equipment which would influence planning of the decontamination and decommissioning of 233-S Building.

During review of the draft procedures and discussion of subsequent program work in the building, it was concluded that portions of work which were to be performed under Activity Specification II at a later date should be included in the initial

*Recent information has been obtained which indicates that some of these estimates of performance and the resulting priority indices are incorrect and need to be adjusted before priorities are finally established.

TABLE 9. Prioritization of Retention Basin Systems

USING THE FOLLOWING CRITERIA WEIGHTS
 OFFSITE HAZARD=24.6 ONSITE HAZARD=40.7 COST=11.6 COMPATIBILITY=23.2

FACILITY NUMBER	FACILITY NAME	STATUS	OFFSITE HAZARD	ONSITE HAZARD	LAYAWAY COST	FUTURE COMPAT- IBILITY	PRIORITY INDEX
107-KW-C	RETENTION BASIN	STANDBY	7	7	8	8	735
107-KW-B	RETENTION BASIN	STANDBY	7	7	8	8	735
107-KW-A	RETENTION BASIN	STANDBY	7	7	8	8	735
107-KE-C	RETENTION-BASIN	STANDBY	7	7	8	8	735
107-KE-B	RETENTION-BASIN	STANDBY	7	7	8	8	735
107-KE-A	RETENTION-BASIN	STANDBY	7	7	8	8	735
107-C-(2)	RETENTION-BASIN	STANDBY	6	6	7	7	635
107-C-(1)	RETENTION-BASIN	STANDBY	6	6	7	7	635
107-F	RETENTION-BASIN	INACTIVE	5	6	5	6	564
116-K-2	V CRIB	INACTIVE	7	1	10	10	561
107-D	RETENTION BASIN	STANDBY	7	4	5	5	509
107-DR	RETENTION-BASIN	INACTIVE	5	5	5	5	500
107-D	RETENTION BASIN	INACTIVE	5	4	5	5	460
107-H	RETENTION BASIN	INACTIVE	5	3	5	6	431
116-K-1	V CRIB	INACTIVE	5	1	5	6	361
116-H-1	V TRENCH	INACTIVE	4	1	5	6	336
116-C-1	V TRENCH	INACTIVE	4	1	5	6	336
116-DR-1	V TRENCH	INACTIVE	4	1	5	5	313
116-F-2	V TRENCH	INACTIVE	3	1	5	6	312
107-F-PIPE	EFFLUENT-LINES	INACTIVE	1	3	1	6	297
116-B-1	V TRENCH	INACTIVE	4	1	4	4	278
107-K-PIPE	EFFLUENT LINES	STANDBY	1	3	1	4	251
116-K-3	OUTFALL-STRUCT	ACTIVE	1	3	2	3	240
107-D/DR PIPE	EFFLUENT-LINE	INACTIVE	1	2	1	5	234
107 B/C-PIPE	EFFLUENT-LINES	STANDBY	1	2	1	5	234
116-DR-2	V TRENCH	INACTIVE	2	1	4	4	229
116-D-5	OUTFALL-STRUCT	ACTIVE	1	3	2	2	216
116-B-7	OUTFALL-STRUCT	ACTIVE	1	3	2	2	216
207-U	RETENTION-BASIN	INACTIVE	2	2	3	1	189
207-S	RETENTION-BASIN	INACTIVE	4	1	2	1	186
207-T	RETENTION-BASIN	ACTIVE	3	1	3	1	173
207-B	RETENTION BASIN	ACTIVE	3	1	3	1	173
107 H-PIPE	EFFLUENT-LINES	INACTIVE	1	1	1	4	170
116-F-8	OUTFALL-STRUCT	INACTIVE	1	2	1	2	164
207-SL	RETENTION-BASIN	ACTIVE	1	2	1	1	141
207-U-PIPE	EFFLUENT-LINE	INACTIVE	1	1	1	1	100
207-T-PIPE	EFFLUENT-LINE	ACTIVE	1	1	1	1	100
207-SL-PIPE	EFFLUENT-LINE	ACTIVE	1	1	1	1	100
207-S-PIPE	EFFLUENT-LINE	INACTIVE	1	1	1	1	100
207-B-PIPE	EFFLUENT-LINE	ACTIVE	1	1	1	1	100
116-H-5	OUTFALL-STRUCT	INACTIVE	1	1	1	1	100
116-DR-5	OUTFALL-STRUCT	INACTIVE	1	1	1	1	100
116-C-4	OUTFALL-STRUCT	STANDBY	1	1	1	1	100
116-B-8	OUTFALL-STRUCT	STANDBY	1	1	1	1	100

field activities. Combining portions of work under Specification II with Specification I into a single concurrent effort provides improved safety and work efficiency and will demonstrate a more typical approach to entry and safe occupancy of an alpha-contaminated facility.

Under the revised plan, the building will be divided into areas of similar levels of contamination and equipment. Each area will be surveyed, radiation removed or fixed, and equipment removed from the building. In this way, relatively safe areas of operation will be established stepwise throughout the facility. This will provide safe change areas and storage of supplies and packaged waste materials within the protective structure of 233-S Building. A complete record will be kept of the time, technique, and effectiveness of each technique used within the areas.

TASK 3 - Exploratory R&D -

K. M. Harmon

Demonstration of Techniques for Stabilization of a Waste Burial Ground -

E. L. Klepper

The demonstration trench has been installed as planned, and the surface has been planted to cheatgrass. The parallel experimental plots have been installed for ARHCO and planted. A gravelled road to the project has been put in to permit year-round access.

REFERENCE

1. D. A. Waite and C. E. Jenkins, "Development of Disposition Criteria Derivation Methodology for Commercial Fuel Cycle Facilities," CONF-750827, Proceedings of the Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities, Idaho Falls, Idaho, August 19-21, 1975.

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6.0 TRANSPORTATION SAFETY STUDIES

L. D. Williams - Project Manager

The objective of the Transportation Safety Studies Program is to develop and use a model for calculating the risks associated with the shipping of radioactive materials.

SUMMARY

A first rough draft of an assessment of the risk in transporting plutonium by train has been completed and is under internal review. The results indicate that the risk spectrum for the shipment of PuO_2 by train is similar to that previously calculated for truck shipment. The risk spectrum for the $\text{Pu}(\text{NO}_3)_4$ solution shipment by rail appears to be about an order of magnitude greater than that previously found for plutonium nitrate shipment by truck.

As assessment of the risk in shipment of chlorine by rail has been initiated. A fault tree indicating the various ways that chlorine can be released during transport has been drawn and other system description information has been gathered.

Work on an assessment of the risk in transporting UF_6 continued. The system description has been completed. Fault trees have been drawn, and a survey questionnaire to obtain human factor data has been produced. Efforts to develop transportation accident environment information for the large UF_6 containers were started.

Work also continued on developing an assessment of the risk in the shipment of low-level beta-gamma wastes. The dose conversion factors and waste dispersion fractions have been defined for the beta-gamma wastes.

TECHNICAL PROGRESS

Rail Shipment of Plutonium

Tentative fault trees and event probabilities for use in assessment of the risk of shipping plutonium by train were reviewed, and an initial run of the risk assessment code HEAD was made. The results of the analysis indicated that the risk spectrum for the shipment of PuO_2 by rail is similar to that previously obtained for shipment by truck.* The risk spectrum for $\text{Pu}(\text{NO}_3)_4$ transport by rail is less than an order of magnitude higher than that for plutonium nitrate shipment by truck.*

An initial rough draft of the assessment of the risk of shipping plutonium by train has been prepared. This report incorporates the results obtained in the initial runs of HEAD. It is currently undergoing internal

*BNWL-1846, An Assessment of the Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck.

review. A draft report will be produced after the internal comments are received and evaluated.

Chlorine Transport Risk Assessment

Work toward development of a system description for assessment of the risk in the shipment of chlorine by rail was initiated. Information on shipping container characteristics, safety features, chlorine dispersion characteristics, chlorine health effects, and accident experience in chlorine shipment by rail has been collected. Information on projected rail shipments of chlorine for the Year 1978 is given in Table 10. The risk assessment will focus on shipments in the 55 and 85-90 ton cars.

An initial chlorine fault tree has been produced. It is currently undergoing internal review for correctness and completeness.

is being considered. Also, the performance of the 2 1/2- and 10-ton cylinders with overpack will be evaluated. Fault trees for use in risk assessment were refined to increase the detail of the analysis. A survey questionnaire to help determine packaging conditions during shipment, for use in assigning probabilities in the fault trees, was completed and will be sent to receivers of UF_6 vessels. A questionnaire to determine the amount of UF_6 being shipped and points of origin and destination will also be included. This will be used to check the accuracy of data used in constructing the system description. Samples of these questionnaires are shown at the end of this section.

Work was begun to define the accident environment for the large UF_6 vessels. An evaluation was started to determine what modifications to

TABLE 10. Chlorine Tank Car Shipments
Projected to 1978

<u>Car Designation</u>	<u>Annual No. of Shipments</u>	<u>Tons of Shipped Chlorine</u>
16 tons	440	9,677
30 tons	2,233	67,010
55 tons	27,537	1,514,557
85-90 tons	22,902	2,015,709

Risk Assessment of UF_6 Shipment by Truck and Rail

The system description of the shipment of UF_6 throughout the United States has been completed. Projected shipment flows, origins, and destinations have been defined. Shipment in four different vessels, two 2 1/2-ton vessels (30A and 30B), the 48X 10-ton vessel, and the 48Y 14-ton vessel,

Sandia's small container impact and fire environments would be needed to make them applicable in the UF_6 risk analysis. Fire and impact failure threshold determinations for the UF_6 vessels have begun. An evaluation of data and methods for estimating the airborne release, and adverse health effects, of UF_6 in postulated transportation accidents has been made.

Beta-Gamma Waste Transport Risk Analysis

Contacts were made with several commercial radioactive waste burial firms to request their cooperation in a survey to determine the condition of radioactive waste shipping packages as received. Initial indications are that all of the companies will cooperate in the survey. The information from the survey will be used to determine event probabilities for use in the beta-gamma waste release fault tree.

Shipping requirements for the beta-gamma shipping system in the early 1980s have been assembled. Waste producing sites have been defined. The mileage between the producing sites and the burial sites has been determined. This information completes the system description portion of the risk assessment.

Dose conversion factors for the isotopes in the radioactive waste

have been calculated. Table 11 gives the dose conversion factors for the different forms of waste. The values are postulated maxima for each of the waste types.

Dispersion characteristics have been developed for each form of waste. Table 12 gives the dispersion fractions that will be used in the assessment. These values will be used in determining the amount of material made airborne in an accident.

Work has been started to evaluate the clean-up costs which would be incurred as the result of an accident involving release of beta-gamma waste.

TABLE 11. Dose Conversion Factors*

<u>Waste Forms</u>	<u>50 Year Dose</u>	
	<u>Dose Conversion Factors</u>	
	<u>rem/Ci</u>	<u>Sec/m³</u>
Resin	630	Lungs
Sludge	630	Lungs
Misc.	8300	Bone

*Personal communication with
W. Kennedy - BNW.

TABLE 12. Dispersion Fractions for Waste*
(Fraction of Radioactive Material
made Airborne)

Waste Form	Accident Involving Fire	Non-Fire Accident
Resin (Loose)	0.11	0.008
Sludge (Concrete)	0.01/hr	0.0088
Misc. Waste (Paper, rags, plastic, etc.)	0.38	0.004

*Personal communication with
S. Sutter - BNW.

SURVEY OF TRANSPORT OF UF₆
ENRICHMENT FACILITIES

A. APPROXIMATE PERCENTAGE OF SUPPLY FROM SOURCES:

1. Allied Chemical (Metropolis, IL)
2. Kerr-McGee (Sequoyah, OK)
3. Other Domestic Enrichment Plants
 - a. Paducah, KY
 - b. Oak Ridge, TN
 - c. Portsmouth, OH
4. Foreign Sources (Only Port of Entry to U.S. is Needed)
5. Other Sources (Please List)

B. TYPE OF TRANSPORT (APPROXIMATE PERCENTAGE OF 1971-1975 TOTAL RECEIVED)

	<u>2 1/2 TON</u>	<u>10 TON</u>	<u>14 TON</u>
1. Rail	_____	_____	_____
2. Truck	_____	_____	_____
3. Piggy-back	_____	_____	_____
4. Other (Please Specify)	_____	_____	_____
TOTAL	100%	100%	100%

- C. Do future plans (next 10 years) include enrichment of reprocessed UF₆ from fuel reprocessing plants?
- D. Number of shipments of enriched UF₆ to foreign customers (1970-1974) and ports of exit from U.S.

SURVEY OF TRANSPORT OF UF₆
FUEL FABRICATION AND/OR PREPARATION FACILITIES

A. APPROXIMATE PERCENTAGE OF SUPPLY FROM SOURCES

1. Gaseous Diffusion Plants
 - a. Paducah, KY
 - b. Oak Ridge, TN
 - c. Portsmouth, OH
2. Natural Enrichment UF₆ - Please List Source
3. Depleted UF₆ (TAILS) - Please List Sources
4. Other Sources - Please List

B. TYPE OF TRANSPORT (APPROXIMATE PERCENTAGE OF 1971-1975 RECEIVED)

	<u>2 1/2 TON</u>	<u>10 TON</u>	<u>14 TON</u>
1. Rail	_____	_____	_____
2. Truck	_____	_____	_____
3. Piggy-back	_____	_____	_____
4. Other (Please Specify)	_____	_____	_____
TOTAL	100%	100%	100%

- C. Do future plans (next 10 years) include facilities to handle larger (10 and 14 ton) UF₆ cylinders if such facilities do not currently exist?

SURVEY OF UF₆ PACKAGING CONDITIONS DURING TRANSPORT

	2 1/2-Ton Model <u>30A</u> <u>30B</u>	10-Ton Model <u>48X</u>	14-Ton Model <u>48Y</u>
A. Packages Received			
1975	_____	_____	_____
1974	_____	_____	_____
1973	_____	_____	_____
1972	_____	_____	_____
1971	_____	_____	_____
B. UF ₆ Cylinder Conditions			
1. Average lifetime of a shipping cylinder (years of use).	_____	_____	_____
2. Do you formally examine and record the condition of received cylinders?	_____	_____	_____
3. Are the H/U ratio and the enrichment verified upon receipt?	_____	_____	_____
4. Number of times H/U ratio exceeded specified limits.	_____	_____	_____
5. Number of times enrichment exceeded specified limits.	_____	_____	_____
6. Number received overfilled.	_____	_____	_____
7. Number received overpressured.	_____	_____	_____
8. Number of cylinders requiring repair or replacement of parts prior to reuse.	_____	_____	_____
9. Number requiring valve replacement.	_____	_____	_____
10. Number requiring plug replacement.	_____	_____	_____
11. Number requiring rethreading.	_____	_____	_____
12. Number removed from service due to deterioration.	_____	_____	_____

For questions contact - J. F. Johnson, Battelle-Northwest, Richland, WA 99352 (509) 946-2862
or FTS 444-7411

	2 1/2-Ton Model		10-Ton Model	14-Ton Model
	30A	30B	48X	48Y
13. Number with plug inadequately tightened (loose or insufficient engaged threads).	_____	_____	_____	_____
14. Number with plug and/or vessel threads partially stripped.	_____	_____	_____	_____
15. Number with plug cross-threaded	_____	_____	_____	_____
16. Number with valve inadequately tightened (loose or insufficient engaged threads).	_____	_____	_____	_____
17. Number with valve and/or vessel threads partially stripped.	_____	_____	_____	_____
18. Number with valve cross-threaded.	_____	_____	_____	_____
19. Number with valve not completely closed.	_____	_____	_____	_____
20. Number with badly corroded welds.	_____	_____	_____	_____
21. Number received without required protective valve cover.	_____	_____	_____	_____
22. Number received with contamination outside cylinder.	_____	_____	_____	_____
23. Number received with plug damaged from rough treatment during handling or shipping. (Example plug bent and fresh metal exposure, etc.)	_____	_____	_____	_____
24. Number received with welds damaged from rough treatment during handling or shipping (example, vessel freshly dented in area of weld resulting in possible partial weld failure, etc.)	_____	_____	_____	_____
C. Overpack Conditions				
1. Number of cylinders with overpack received.	_____	_____	_____	_____
2. Number with gross damage to overpack.	_____	_____	_____	_____
3. Number received without sealing gasket.	_____	_____	_____	_____
4. Number with deteriorated gasket.	_____	_____	_____	_____
5. Number with closure bolts inadequately tightened.	_____	_____	_____	_____
6. Number with missing closure bolts.	_____	_____	_____	_____
7. Number with damaged overpack mating surface.	_____	_____	_____	_____
8. Number received without required overpack.	_____	_____	_____	_____

For questions, contact - J. F. Johnson, Battelle-Northwest, Richland, WA 99352 (509) 946-2862
or FTS 444-7411

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7.0 SAFETY AND ECONOMIC STUDY OF SPECIAL TRAINS

L. D. Williams - Project Manager

The objective of this program is to compare the safety and economics of special trains with those of regular trains for the shipment of nuclear fuel cycle material.

SUMMARY

Activity on this program began in November 1975. A program plan was established and work on the first phase of the program begun. The identification and characterization of commercial power reactor spent fuel origin and destination sites and projected quantities shipped for 1976, 1981 and 1986 have been completed.

TECHNICAL PROGRESS

As the nuclear power industry grows, there will be a relatively large increase in the number and frequency of spent fuel shipments. Due to concern about the safety of these shipments, the railroads have proposed requirements that these shipments be made on special trains having limitations such as: that the maximum speed be 35 mph and that when a special train and another train pass, one train must be stationary. Currently the overall effects on safety and economics of the use of special trains are largely unknown. This quarter a program plan was established and work initiated toward evaluating these effects.

The first phase of the program is the identification and characterization of spent fuel transportation network. This phase is divided into two

parts. One is identification of origins and destinations of spent fuel in 1976, 1981 and 1986. The other is determination of routes between the specified origin-destination points and the relevant variables associated with these routes.

The identification and characterization of commercial power reactor spent fuel origin and destination sites and the quantities shipped have been completed. This information is shown tabulated in Tables 13 through 14. A total of 63 origin-destination pairs have resulted from this phase of the study. Some refinement of these combinations will follow as additional data is obtained. The total rail shipments of commercial power reactor spent fuel have been estimated at 100 MT for 1976, 1450 MT for 1981, and 2250 MT for 1986.

Identification and characterization of the railroad network for these origin-destination pairs should be completed next quarter. Results expected from this part of the study are description, performance, and cost of railroad service for regular and special trains.

The safety analysis phase of the study is in the process of analyzing the railroad accident environment. Train accident data for the years

1972 and 1973 are being analyzed for information about frequency--severity relationships in freight train accidents. The data are on a tape provided by the Federal Railroad Administration which contains a record of 52,770 train and train-service accidents. Of these, 9890 are reportable train accidents involving freight trains, and 302 are reportable train accidents involving passenger trains. The remainder are reportable train accidents involving other types of trains and train-service accidents. Train accidents are categorized as collisions, derailments and "other" accidents in which there is a least \$750 damage to railroad property.

Relevant information on the accident tape includes: train speed; class and subclass of accident; accident cause; dollar damage to equipment and to track and roadbed; casualties; and number of train cars.

The following tables, 13 through 16, summarize postulated spent fuels shipment activity from reactor sites to the three identified reprocessing plants and the G.E. storage site. It should be noted that quantities shipped are less than the quantities discharged from the reactors. The limited reprocessing capacity through the early 1980s will require extended storage at some reactors.

The selection of plants and their shipments to the reprocessing sites was made on the basis of older plants having priority, available reprocessing sites which accepted shipments, and availability of railroad spurs which permitted use of rail casks.

TABLE 13. Summary of Spent Fuel Shipments

	1976 MTU/% Rail	1981 MTU/% Rail	1986 MTU/% Rail
AGNS (Barnwell, SC)	0/0	1500/90%	1500/90%
NFS (West Valley, NY)	0/0	0/0	750/01%
EXXON (Oak Ridge, TN)	0/0	0/0	1500/60%
GE (Morris, IL)	150/67%	150/67%	0/0
Total MTU by Rail	100	1450	2253
Number of Shipments*	32	369	523

*Assumes shipment in the following casks:

AGNS - NLI 10 PWR/24 BWR cask
 EXXON - 10 PWR/24 BWR EXXON cask
 NFS - NFS 2 cask (1.2 MTU capacity)
 GE - IF-300 cask (~ 3.25 MTU capacity)

TABLE 14. Plants Shipping Spent Fuel in Calendar Year 1976

	Plant	Shipments	Origin	Rail Service	Destination
1.	Dresden 1	3	Morris, IL		Morris, IL
2.	Dresden 2	7	Morris, IL		Morris, IL
3.	Dresden 3	7	Morris, IL		Morris, IL
4.	Browns Ferry 1	10	Athens, AL	9 miles (truck) to Southern or L & N	Morris, IL
5.	Monticello	5	Monticello, MN	Burlington Northern	Morris, IL

TABLE 15. Plants Shipping Spent Fuel in Calendar Year 1981

	Plant	Shipments	Origin	Mi	Rail Service	Destination
1.	Dresden 1, 2, 3	15	Morris, IL		Direct	Morris, IL
2.	Hatch 1, 2	16	Baxley, GA	11	Southern	Morris, IL
3.	Cooper	8	Brownville, NB	2	Chicago, Burlington & Quincy	Morris, IL
4.	Monticello	5	Monticello, MN	3	Burlington Northern	Morris, IL
5.	Arkansas 1, 2	11	Russellville, AR	6	Missouri Pacific	Barnwell, SC
6.	Beaver Valley 1	5	Shippingport, PA	1	New Cumberland & Pittsburg	Barnwell, SC
7.	Brunswick 1, 2	10	Southport, NC	2	Seaboard Coastline	Barnwell, SC
8.	Cook 1, 2	11	Benton Harbor, MI	11	Chesapeake & Ohio	Barnwell, SC
9.	Crystal River	5	Tampa, FL	70	Atlantic Coast Line	Barnwell, SC
10.	Davis Besse 1	5	Oak Harbor, OH	6	Norfolk & Western	Barnwell, SC
11.	Duane Arnold 1	4	Cedar Rapids, IA	8	Chicago, Rock Island & Pacific	Barnwell, SC
12.	Farley 1, 2	10	Dothan, AL	7	Seaboard Coastline	Barnwell, SC
13.	Fitzpatrick 1	6	Oswego, NY	7	Penn Central	Barnwell, SC
14.	Fort Calhoun 1	3	Blair, NB	4	Chicago & Northwestern	Barnwell, SC
15.	Ginna 1	3	Rochester, NY	16	Penn Central	Barnwell, SC
16.	Maine Yankee	6	Portland, ME	45	Maine Central	Barnwell, SC
17.	McGuire 1, 2	11	Charlotte, NC	11	Seaboard Coastline	Barnwell, SC
18.	Millstone 1, 2, 3	16	Hartford, CT	40	NY, NH, HRR to Penn Central	Barnwell, SC
19.	North Anna 1, 2	9	Mineral, VA	10	Chesapeake & Ohio	Barnwell, SC
20.	Oyster Creek 1	6	Toms River, NJ	9	Central RR of New Jersey	Barnwell, SC
21.	Palisades	6	South Haven, MI	5	Chesapeake & Ohio	Barnwell, SC
22.	Peach Bottom 2, 3	15	Peach Bottom, PA	1	Maryland & Pennsylvania	Barnwell, SC
23.	Prairie Island 1, 2	6	Redwing, MN	6	Chicago, Milw. St. Paul & Pacific	Barnwell, SC
24.	Quad Cities 1, 2	11	Cordova, IL	3	Chicago, Milw. St. Paul & Pacific	Barnwell, SC
25.	Rancho Seco	6	Sacramento, CA	25	South Pacific	Barnwell, SC
26.	Robinson 2	5	Hartsville, SC	5	Seaboard Coastline	Barnwell, SC
27.	San Onofre 1	4	San Clemente, CA	4	At. Topeka & Santa Fe	Barnwell, SC
28.	Sequoyah 1, 2	11	Chattanooga, TN	12	Southern	Barnwell, SC
29.	Three Mile Island 1, 2	11	Middletown, PA	5	Penn Central	Barnwell, SC
30.	Trojan	6	Rainier, OR	4	Burlington Northern	Barnwell, SC
31.	Virgil Summer 1	6	Jenkinsville, WI	3	Southern	Barnwell, SC
32.	Vermont Yankee	4	Vernon, VT	1	Central Vermont	Barnwell, SC
33.	Watts Bar 1, 2	11	Spring City, TN	10	Southern	Barnwell, SC
34.	WPPSS 2	8	Richland, WA	15	Chicago, Milw. St. Paul & Pacific	Barnwell, SC
35.	Yankee	2	Rowe, MA	3	Boston & Maine	Barnwell, SC
36.	Zion 1, 2	11	Zion, Chicago, IL	40	Chicago & Northwestern	Barnwell, SC
37.	Oconee 1, 2, 3*	23	Seneca, SC	8	5 mi (truck) to Southern	Barnwell, SC
38.	Turkey Point 3, 4*	13	FL. City, Miami, FL	25	9 mi (truck) to Florida East Coast	Barnwell, SC
39.	St. Lucie 1, 2*	15	Ft. Pierce, FL		9 mi (truck) to Florida East Coast	Barnwell, SC
40.	Browns Ferry 1, 2, 3*	30	Athens, AL	10	9 mi (truck) to Southern or L & N	Barnwell, SC

Note: All shipments to Morris, IL made in intermodal 7/18 casks with 3.25 MT capacity.
 All shipments to Barnwell, SC made in rail 10/24 casks with 4.52 MT capacity except for those marked with (*), which are made in 7/18 intermodal casks.

TABLE 16. Plants Shipping Spent Fuel in Calendar Year 1986

	Plant	Shipments	Origin	Mi	Rail Service	Destination
1.	Arkansas 1, 2	11	Russellville, AR	6	Missouri Pacific	Oak Ridge, TN**
2.	Bailly	4	Gary, IN	10	Chicago South Shore & South Bend	Oak Ridge, TN
3.	Beaver Valley 1, 2	9	Shippingport, PA	1	New Cumberland & Pittsburg	Barnwell, SC
4.	Braidwood 1, 2	11	Braidwood, IL		Illinois Central Gulf	Oak Ridge, TN
5.	Brunswick 1, 2	11	Southport, NC	2	Seaboard Coastline	Barnwell, SC
6.	Comanche Peak 1, 2	11	Somervell Co., TX		At. Topeka & Santa Fe	Oak Ridge, TN**
7.	Cook 1, 2	11	Benton Harbor, MI	11	Chesapeake & Ohio	Barnwell, SC
8.	Cooper	6	Brownville, NB	2	Chicago, Burlington & Quincy	Oak Ridge, TN
9.	Crystal River	6	Red Level, Tampa, FL	70	Atlantic Coast Line	Barnwell, SC
10.	Davis Besse 1, 2	11	Oak Harbor, OH	6	Norfolk & Western	Barnwell, SC
11.	Duane Arnold 1	4	Cedar Rapids, IA	8	Chicago, Rock Island & Pacific	Barnwell, SC
12.	Enrico Fermi 2	7	Monroe, MI	8	New York Central & Detroit & Toledo	Oak Ridge, TN
13.	Farley 1, 2	10	Dothan, AL	7	Seaboard Coastline	Barnwell, SC
14.	Fort Calhoun 1, 2	8	Blair, NB	4	Chicago & Northwestern	Barnwell, SC
15.	Fitzpatrick 1	6	Oswego, NY	7	Penn Central	Barnwell, SC
16.	Forked River 1	7	Toms River, NJ	9	Central RR of New Jersey	Oak Ridge, TN
17.	Ginna 1	3	Rochester, NY	16	Penn Central	Barnwell, SC
18.	Hatch 1, 2	12	Baxley, GA	11	Southern	Oak Ridge, TN**
19.	Hope Creek 1, 2	15	Bordentown, NJ	3	Penn Central	Oak Ridge, TN
20.	LaSalle 1, 2	14	Seneca, IL	5	At. Topeka & Santa Fe	Oak Ridge, TN
21.	Limerick 1, 2	15	Pottstown, PA	2	Reading	Oak Ridge, TN
22.	Maine Yankee	6	Portland, MA	45	Maine Central	Barnwell, SC
23.	McGuire 1, 2	11	Charlotte, NC	11	Seaboard Coastline	Barnwell, SC
24.	Midland 1, 2	11	Midland, MI		Chesapeake & Ohio	Oak Ridge, TN
25.	Millstone 1, 2, 3	16	Hartford, CT	40	NY, NH, HRR to Penn. Central	Barnwell, SC
26.	Monticello	5	Monticello, MN	3	Burlington Northern	Oak Ridge, TN**
27.	North Anna 1, 2, 3, 4	20	Mineral, VA	10	Chesapeake & Ohio	Barnwell, SC
28.	Oconee 1, 2, 3*	23	Seneca, SC	8	5 mi (truck) to Southern	Barnwell, SC
29.	Oyster Creek 1	6	Toms River, NJ	9	Central RR of New Jersey	Barnwell, SC
30.	Palisades	6	South Haven, MI	5	Chesapeake & Ohio	Barnwell, SC
31.	Peach Bottom 2, 3	13	Peach Bottom, PA	1	Maryland & Pennsylvania	Barnwell, SC
32.	Prairie Island 1, 2	6	Redwing, MN	6	Chicago, Milw. St. Paul & Pacific	Oak Ridge, TN**
33.	Quad Cities 1, 2	11	Cordova, IL	3	Chicago, Milw. St. Paul & Pacific	Oak Ridge, TN**
34.	Rancho Seco	5	Sacramento, CA	25	Southern Pacific	Oak Ridge, TN**
35.	Robinson 2	5	Hartsville, SC	5	Seaboard Coastline	Barnwell, SC
36.	San Onofre 1, 2, 3	16	San Clemente, CA	4	At. Topeka & Santa Fe	Barnwell, SC
37.	Sequoyah 1, 2	11	Chattanooga, TN	12	Southern	Barnwell, SC
38.	Susquehanna 1, 2	14	Shickshinny, PA	5	Erie-Lakawanna	Oak Ridge, TN
39.	Three Mile Island 1, 2	11	Middletown, PA	5	Penn Central	Barnwell, SC
40.	Trojan	6	Rainier, OR	4	Burlington Northern	Oak Ridge, TN**
41.	Virgil Summer	6	Jenkinsville, WI	3	Southern	Barnwell, SC
42.	Vermont Yankee	4	Vernon, VT	1	Central Vermont	Barnwell, SC
43.	Waterford 3	6	Norco, LA	5	Texas & Pacific	Oak Ridge, TN
44.	Watts Bar 1, 2	11	Spring City, TN		Southern	Oak Ridge, TN**
45.	WPPSS 1, 2, 3, 4, 5	32	Richland, WA	15	Chicago, Milw. St. Paul & Pacific	Barnwell, SC
46.	Zion 1, 2	11	Zion, Chicago, IL	40	Chicago & Northwestern	Barnwell, SC
47.	Turkey Point 3, 4*	13	FL City, Miami, FL	25	9 mi (truck) to Florida East Coast	Barnwell, SC
48.	St. Lucie 1, 2*	14	Ft. Pierce, FL		9 mi (truck) to Florida East Coast	Barnwell, SC
49.	Browns Ferry 1, 2, 3*	30	Athens, AL	10	9 mi (truck) to Southern	Barnwell, SC
50.	Humboldt Bay***	1	Eureka, CA	1	Northwestern Pacific	Barnwell, SC
51.	Big Rock Point***	1	Charlevoix, MI	3	Chesapeake & Ohio	West Valley, NY

* Requires truck/rail shipment using 7/18 casks with 3.25 MT capacity. All others use casks with 4.52 MT capacity.

** Shipments were made to Barnwell in 1981.

*** Shipments from Humboldt Bay and Big Rock Point use NFS rail car holding 10 elements.

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8.0 DEVELOPMENT OF HIGH-LEVEL WASTE SHIPPING CASK MODELS

L. D. Williams - Project Manager

The objective of this program is to conceptually design and fabricate models of a cask for transporting high-level waste by railroad.

SUMMARY

Two types of high-level radioactive waste shipping cask models will be fabricated in this program: 1) a complete H0-gage rail car and cask assembly (1:87 scale) and 2) a table model of the cask alone for use in portable displays. The table model will be about 1 ft in diameter by 2 ft long and have a cutaway section showing the waste canister inside. Graphics will also be provided to further illustrate the concept developed and to relate the waste products conveyed to the reactor fuel cycle.

Initial design guidelines for the models have been established.

TECHNICAL PROGRESS

The cask model program is a new program that began midway through the present quarter. Preliminary work has concentrated on outlining design constraints and determining the relevancy of spent fuel cask design parameters to high-level waste casks. The guidelines below are tentatively established for the conceptual design:

- The waste form will be glass.
- Waste canisters will be 1 ft in diameter and 10 ft long.

- The age of the waste at the time of shipment will be at least 5 years.
- The surface temperature of the canister during normal transport should not exceed 700°F.
- Total cask assembly weight will be between 200,000 and 250,000 lb (100 to 125 tons) due to standard rail car limitations.
- The cask will be oriented horizontally during shipment.

Preliminary heat transfer analyses have been completed to determine the surface temperature of the canisters, the temperature in the shielding, and the external surface temperature of the cask. The thermal analysis assumed no auxiliary cooling. The cask configuration used for the analysis included an inner steel cylinder 5 ft in diameter surrounded by 8 in. of lead. A 2-in.-thick structural steel shell with 4-in.-long cooling fins spaced 4 in. apart enclosed the entire assembly. Two 0.060-in. internal air gaps were assumed, one at the canister surface and the other between the inner steel cylinder and the lead shielding. Ambient external temperature was assumed to be 120°F. The results of the analyses are shown in the attached Figure 4.

FIGURE 4. Results of Preliminary Heat Transfer Analyses

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