# ENVIRONMENTAL RADIATION PROTECTION FOR NUCLEAR POWER OPERATIONS

PROPOSED STANDARDS

[40 CFR 190]

SUPPLEMENTARY INFORMATION

JANUARY 5, 1976

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$$\left[40 \text{ CFR } 190\right]$$

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PREFACE

As a result of the review of comments received on these proposed
environmental radiation protection standards for normal operations of
activities in the uranium fuel cycle, the Agency has identified a number
of areas in which additional information would be desirable in order to
provide a reasonable basis for discussion and comment on this proposed
rulemaking at the public hearing scheduled for February 17, 1976. This
material has been developed to supplement that contained in the notice
proposing these standards (40 FR 23420), as well as the draft environ-
mental statement and technical reports made available at that time. It
does not constitute a complete response to comments, since the public
record is still open. Modifications of the original proposal made as
the result of comments received and a complete response to comments will
be contained in the final environmental statment and notice of final
rulemaking, which will reflect all the information received, including
that developed at public hearings.

Three categories of additional information are contained in this
Supplement. The first includes an extended discussion of the Agency's
intent regarding implementation of these proposed standards, and further
elucidation of the basis used by the Agency for assessing the potential
health impact of exposure to ionizing radiation. The second consists of
technical discussions of several areas not covered or addressed only
briefly by the original material. This includes consideration of
multiple reactors on a single site, the nuclear energy center concept,
transuranic effluents resulting from recycled uranium, and nitrogen-16
skyshine doses and control at BWR's. Finally, in two areas, fuel

reprocessing and milling, considerable additional technical material has become available concerning control methods since the original documentation was prepared. Although the proposed standards reflected this information, the technical documents accompanying the proposal did not. Surveys based on this new information complete this collection of additional materials.

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IMPLEMENTATION OF AND VERIFICATION OF

COMPLIANCE WITH THE PROPOSED STANDARDS

# IMPLEMENTATION OF AND VERIFICATION OF COMPLIANCE WITH THE PROPOSED STANDARDS

## Introduction

As pointed out in the notice proposing these standards, the primary responsibility for implementing and assuring compliance with EPA standards for environmental radiation from nuclear power rests with the Nuclear Regulatory Commission (NRC) and, in certain cases, "Agreement States" operating within NRC regulations. Thus, although EPA must consider the practicality of implementing its standards, it would clearly be inappropriate for the Agency to specify the detailed procedures to be followed. On the other hand, it is important that the Agency clearly, spell out what it would consider to be an appropriate implementation, as well as ones which are overly restrictive or inadequate, so as to provide guidance to the NRC for its development of the detailed regulations (and modifications of existing regulations) required. The following comments are intended in the sense of such guidance, as to the the Agency's intent, therefore, and should not be interpreted as literal dictates of the regulations required to implement these standards. That responsibility rests with the NRC, and will have to be worked out by the NRC through detailed interaction with the affected components of industry, with timely consultation by NRC with EPA as to the appropriateness of any proposed implementing regulations, particularly in the event that difficulties develop.

A similar situation obtains with respect to verification of compliance. Enforcement authorities reside in NRC, not EPA. EPA expects that the NRC will adequately assure compliance, and EPA's own "compliance" activities will consists principally of the review of the performance, as reported by NRC, of fuel cycle facilities and of any variances permitted by NRC. As required, EPA will in the future provide NRC with guidance on the adequacy of its compliance and variance posture with respect to these environmental standards.

## Operational vs. Pre-Operational Application of the Standards

An important consideration relative to these standards is the NRC's continuing development of design and operating guidance, codified in 10CFR50, which implements the Federal Radiation Guidance that exposures of the public be maintained as low "as practicable" (25 FR 4402). The Commission has already issued such guidance for single light-water-cooled power reactors and has underway similar guidance for fuel reprocessing, milling, and fuel fabrication facilities. The Agency has determined that the guidance issued thus far for light-water-cooled reactors provides adequate assurance of compliance (unless the NRC finds that extreme extenuating circumstances exist for a specific site) for sites containing up to at least five such power reactors. Additional guidance may be required in the future, as noted by the Commission in its opinion filed with 10 CFR 50, Appendix I, for sites containing larger numbers of facilities.

These standards will supercede, for the nuclear power industry, the Federal Radiation Guides codified in 10 CFR 20 as limiting concentrations in air and water at unrestricted locations. Just as the development of the guidance expressed by Appendix I to 10 CFR 50 took place within the limitations specified by those standards, the development of future 10 CFR 50 guidance will now take place within the limits specified by these standards. However, it is not anticipated that the disparity between standards and guidance will, in general (but not always), be nearly so great as formerly. For example, at fuel reprocessing sites, a large portion of the thyroid individual dose standard could be taken up by new 10 CFR 50 guidance (whereas zero dose may be postulated through liquid pathways due to the absence of any liquid discharges). It is thus not the intent of the Agency that the standards for dose be "apportioned" to various operations of the fuel cycle. They apply equally and in full to doses from any operation or combination of operations in the cycle, and it is not anticipated that doses from multiple sites will be either common or significant. In the few instances where overlap of significance could occur this should be dealt with on a site-specific basis -- not generically through apportionment.

It is particularly important to recognize that the standards apply only to doses received by individuals and quantities of radioactive materials released to the environment from operating facilities. This situation is in contrast to design guidance set forth, for example, by Appendix I to 10 CFR 50 for light-water-cooled power reactors, which applies to pre-operational considerations, such as licensing for

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construction of nuclear facilities. While such guidance is useful for providing the basis for concluding that such facilities can be expected to conform to standards which apply to actual operations, it is not a substitute for such standards.

Consideration of the adequacy of control measures at facilities during pre-operational stages with respect to these standards should be limited to a finding, either for specific sites, or on a generic basis, as appropriate, that the facility has provided or has available to it adequate means to provide reasonable assurance that these standards can be satisfied during actual operations. Such means may include the provision of cleanup controls on discharge streams, the ability to modify, if necessary, its mode of operation to mitigate environmental, discharges, or methods which interrupt exposure pathways in the environment. The important point is that the standards specify maximum doses to real individuals and maximum quantities of certain materials actually delivered or discharged to the environment, not the specific design parameters of individual facilities. Thus, for example, it is the Agency's view that conformance to Appendix I by a planned reactor on a site containing up to five such facilities (unless extremely unusual combinations of liquid and air pathways of exposure are actually present and are expected to be simultaneously intercepted by real individuals) should constitute _de facto_ demonstration to the NRC that a reasonable expectation exists that these standards can be satisfied in actual operation. The Agency will, in the course of its continuing review of Environmental Statements, identify any situations for which it believes

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that such an expectation has not been adequately justified. A more detailed exposition of some areas meriting in-depth discussion of the Agency's view of an adequate demonstration of reasonable expectation of compliance, such as for adjacent sites, minor releases of specifically limited radionuclides from fuel cycle facilities, doses from windblown material originating from mill sites, and transportation-related doses, is provided below.

## Models for Operational Application of the Standards

a) Limits on doses to individuals.

Conformance to the standards should be measured using the most reasonable and, as required, realistic means available. Thus, in the case of dose to the thyroid, measurement of the radioiodine content of milk at the nearest farm, coupled with a determination of the milk consumption habits of the residents, would constitute a reasonable basis for a final determination of noncompliance. Conversely, calculations based on observed releases and meteorology should generally provide the basis for a routine finding of compliance. Sites failing this test would merit progressively more detailed study, leading finally to the above-described (or a comparable) determination of noncompliance (or compliance).

In the case of potential doses to the whole-body and other organs a similar sequence of compliance verification methods is available. The Agency believes that it may be presumed that existing models for

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calculation of exposure fields due to gaseous and liquid releases, using measured data on quantities released, local meteorology, and stream-flow characteristics, are adequately conservative to serve as the basis for verification of compliance with these standards. If reason exists to believe, based on use of such source term measurements and models, that noncompliance may exist at a particular site, than more detailed field measurements may be employed (or, of course, the facility could reduce its emissions to achieve model-based compliance).

In a very few special situations when two or more sites are in close proximity, it may be necessary for the regulatory agency to make allowance for contributions from several sites in order to assure compliance with the standards at locations intermediate between such sites. For sites as close as a few miles from each other overlapping contributions of as much as 10 to 20% may be possible. The NRC should make the necessary adjustments in the individual technical specifications of facilities at such sites to provide reasonable assurance of compliance. However, in the vast majority of situations the sum of all reasonably possible contributions from all sources other than the immediately adjacent site will be small compared to these standards, and should be ignored in assessing compliance. It would not be reasonable to attempt to incorporate into compliance assessment doses which are small fractions of the uncertainties associated with determination of doses from the primary source of exposure.

A number of potential difficulties exist regarding implementation of the standards at mill sites. Gamma surveys in the vicinity of some

existing mill tailings piles show values ranging up to several hundred mrem/yr in situations where it is logical to assume that these elevated gamma radiation levels are the result of windblown tailings. Although the measurement of 25 mrem/yr increments in such dose rates is possible, rigorous measurement techniques would be required to identify locations where new depositions of windblown particulates elevate pre-existing local levels by 25 mrem/yr. Furthermore, because of the projected 20-year operational lifetime of a typical mill and the assumed additive impact of new depositions, 1/20 of 25 mrem/yr, or approximately one mrem/yr, would have to be measured if the standard were to be implemented by a regulation based on verification on an annual, incremental basis. This would be unreasonable, since one mrem/yr is small compared to uncertainties in natural gamma-ray background levels.

A recent engineering survey report developed for the Nuclear Regulatory Commission (ORNL-TM-4903, Volume 1) provides an estimation of the relative ratio of the respirable particles (<10μ) to larger particles (10-80μ) blown off the tailings beach of a well-managed tailings impoundment system. This ratio averages about one and varies from 0.4 to 1.4 depending on specifics of the milling process and other variables. It can be estimated, therefore, that one millicurie/yr of insoluble 0-10μ particles removed from a typical pile by wind could deliver a dose equivalent of approximately one mrem/yr to the lungs of a person living one kilometer downwind of the pile. At the same time, one millicurie/yr of 10-80μ particles might be deposited in a ring one-half to one kilometer from a pile, yielding a surface contamination level of about 3

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$nCi/m^2$. This would result in a gamma-ray exposure level of about 10 $\mu$rem/yr. After 20 years of operations, each contributing to surface contamination at such a rate, this exposure might increase to as much as approximately 0.2 mrem/yr.

Accordingly, the critical exposure pathway for windblown tailings is most likely to be to the lungs through the direct inhalation of radioactive tailings; and if this source of exposure is controlled, direct whole-body gamma exposure from windblown tailings will also be controlled to a considerably greater degree.

It does not appear at this time to be practical to measure the annual release of radionuclides from operational tailings piles to the air pathway. However, it is practical and reasonable to reduce these releases to very small values (<1 mCi/yr) by application of control measures that will insure that maximum doses to individuals in the vicinity of tailings piles are well within the standards. These measures include back-filling of exposed tailings, keeping tailings under water, and spraying any tailings "beaches" that develop with chemical binders to prevent blowing. In practical terms, the standards should be implemented with regard to operational tailings piles by requiring proper and reasonable dust control measures and by permanent stabilization following termination of active milling operations.

It should be noted that the standards apply only to annual doses delivered as the result of discharges of radioactive materials beginning two years following the promulgation date. They do not apply to doses resulting from discharges before this date. Decontamination of areas

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contaminated by*windblown tailings from and management of tailings piles on previously abandoned mill sites are not covered by and are therefore not required by this standard.

At a fuel reprocessing or a multi-unit reactor site the number of shipments of radioactive materials per year in and out of the site could reach several thousand. However, even for this large of number of shipments, doses to nearby individuals under present Department of Transportation regulations would not reach one mrem/yr, if they are located, on the average, more than a few tens of meters from the shipping route, and if the vehicles involved remain in motion while in the vicinity of the site. Implementation of the standard does not require, therefore, modification of existing packaging and shielding requirements. It probably will be necessary, however, to reguire guaranteed non-stop shipments (a service which is presently obtainable from the transportation industry) to avoid buildup of doses to bystanders at habitual stopping places, or to provide restricted access areas for layovers. It should be noted that the standards do not apply to transportation personnel while they are engaged in handling shipments; such exposure is considered to fall in the category of occupational exposure.

b) Limits on quantities of specific radionuclides released.

Implementation of the nuclide-specific limits on releases of long-lived materials will require a determination by the NRC of the operating decontamination factors that must be achieved at locations that are the

principle potential sources of environmental releases of these materials.

In order to make such a determination it will be necessary to

characterize before 1983, except in the case of transuranics, the maximum

average values of environmental releases of these materials from minor

classes of sources to be permitted essentially unrestricted release

(e.g., krypton-85, iodine-129, and transuranic releases from power

reactors or fuel fabrication facilities). Following this, compliance

should consist of verification that the appropriate decontamination

factors are being realized through frequent inplant measurements at the

principle potential sources reported on a routine basis.

Monitoring of the DF's achieved by inplant control systems for the

three types of radionuclides specifically limited by the standards

appears to be readily achievable using conventional monitoring techniques

and analytical procedures, and such measurements appear to be provided

for at the one facility approaching operational status. Flow-through

ionization chambers are capable of measurements of krypton-85 at

concentratons of less than 1 pCi/cm$^3$, a concentration 1000 times lower

than that corresponding to the standard for a typical stack effluent

volume. Similarly, x-ray spectrometry is capable of sensitivities of the

order of 1 pCi for iodine-129; at 10% of the proposed limit a charcoal

sample of stack effluent would accumulate, for a 10 minute sample of 0.2%

of the stream, 1000 pCi. Finally, gas-flow proportional counters, using

24-hour filter samples (collected on 0.1% of the gas stream) would

exhibit detection limits at least 1000 times smaller than activities

corresponding to the standard. Periodic confirmation of the isotopic distribution of transuranics would also be necessary.

It should not be necessary to routinely monitor minor releases of these materials from minor classes of sources, once these have been properly characterized as such, unless normal monitoring of general releases discloses that an unusual situation exists which indicates that normal "de minimus" releases of these materials may be being exceeded. Such an occurrence would, presumably, not constitute a "normal" release and investigation and correction would be warranted in any case.

c) The variance provision.

It is not anticipated that utilization of the variance provision of the standards is likely to be either required or appropriate for any facility other than a power reactor in the foreseeable future. That is not to say that it would be inappropriate to use the variance provision if circumstances warranted, but that such circumstances appear unlikely. On the other hand, it is quite possible that a power emergency, either local, regional, or national, could occur, and that continued production of power by a reactor experiencing higher than normal releases would be in the public interest.

In proposing these standards the Agency purposely did not specify detailed procedures to be followed to obtain a variance, since these should be developed by the NRC with opportunity provided for the views of the interested public and the industry to be heard. The Agency does, however, have some general views on the implementation of this provision.

First, the use of the variance should be predicated upon a demonstrable public need for power, and not on the needs of a utility, as, for example, the inconvenience of scheduling a repair to a control or a fuel reloading. Second, the granting of a variance should be publicly announced, with prior notification of the Agency, and include a brief preliminary assessment of the extent of the excess exposure and releases anticipated, the anticipated duration of the variance, the reason for the excess release, and the reason for granting the variance. Finally, after the variance has terminated, a final assessment of each of the above factors should be issued promptly.

In general it is anticipated, based upon past experience, that when a facility is approaching a condition in which excess releases are possible that normal monitoring and reporting of facility releases will provide more than adequate forewarning to permit timely consideration by NRC of the need for a variance. However, in order to provide for quick response in the case of a sudden power emergency, it may be desirable for the NRC to establish some basic criteria for semi-automatic invocation of a temporary variance under such circumstances. Such criteria would have to be limited, at a minimum, by considerations such as conformance with NRC's safety requirements and FRC occupational exposure limits, limitations which are not affected by these standards.

## Implementing Regulations

A number of regulations or regulatory actions are affected by these standards, as the above discussion of implementation indicates. These include:

1)  10CFR20 - Modify, to reflect, by reference, that 40CFR190 supercedes for normal releases from uranium fuel cycle operations.

2)  10CFR50, Appendix I - Modify to indicate that additional requirements may be required for sites containing more than five light-water-cooled reactors, or, if the NRC so determines, in other special cases.

3)  Review license conditions for fuel cycle facilities, other than light-water-cooled reactors conforming to Appendix I, for conformance to 40CFR190.

4)  Determine whether any sites exist which are close enough to other sites to receive substantial contributions to dose from such sites, and make any necessary modifications of technical specifications in such cases (the Point Beach and Kewaunee sites appear to be the only such potential case presently in existence).

5)  Determine the apportionment to be made for unrestricted release (relative to 40CFR190) of krypton-85, iodine-129, and alpha-emitting transuranics of half-life greater than one year at fuel cycle facilities not major sources of emissions of these nuclides, and determine the decontamination factors required at major sources.

6) Establish criteria, as required, for granting of variances under power emergency conditions, and for establishing public need for orderly delivery of electrical power.

7) Establish, where necessary, requirements on transportation of nuclear wastes and spent fuel to prevent layovers in areas to which public access is possible.

Several regulatory activities already required by existing NRC regulations or underway are also relevant to implementation of these standards. These include:

8) Continuing development of ALAP guidance for fuel cycle activities other than light-water-cooled reactors.

9) Definition of regulatory models for doses to individuals near fuel cycle operations.

10) Definition of "temporary and unusual operating conditions" for implementation of limiting conditions for operation under Appendix I to 10 CFR 50.

The most significant efforts required, of these that are not already required or committed, are items 3), 5), 6), and 7). These concern directly the implementation of the standards, the balance are either minor codifications of the standards into existing regulations, or represent reflection of the existance of these standards into existing ongoing efforts.

## EPA Verification of Compliance

The Agency will assess compliance with these standards through its review of NRC implementing regulations, review of operating data supplied to the NRC by licensees, and review of any variances issued by NRC. Supporting activities will include the Agency's continuing review of · draft and final environmental statements for all fuel cycle facilities, field studies at selected fuel cycle facilities, and assistance to the NRC, when necessary, through field measurements in cases of possible noncompliance.

Under general NEPA and FRC authorities, the Agency routinely reviews and comments on all NRC regulations, including 10 CFR 50 guidance and regulatory guides, pertaining to environmental releases and exposures of the public due to nuclear fuel cycle operations. In the future, this review will also include consideration of the implementation of these standards. This review will encompass, among others, the appropriateness of design basis assumptions, environmental transport models, dose conversion assumptions, environmental monitoring and reporting requirements, and, finally, operating compliance requirements. The Agency will not, however, routinely review technical specifications or other license requirements pertaining to individual licensees.

The Agency also maintains a continuing review of the state of the environment with respect to contamination by radionuclides and doses to the public, including contributions from fuel cycle sources. Beginning this year, the results of this review will be published annually. This report will depend, for fuel cycle sources, primarily upon data collected

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by the NRC.  The Agency has requested that the NRC supply this information in sufficient detail to permit reasonably detailed annual assessments of the exposures of members of the public and releases to the environment at fuel cycle facilities.  Unfortunately, it will apparently be some time before data for all fuel cycle facilities can be made available in a suitable form.

EPA's review of draft and final impact statements for individual fuel cycle facilities will serve to allow EPA to identify to NRC situations in which it believes future compliance, when the facility is completed, may be questionable.  However, such findings will remain advisory, as in the past, since responsibility for compliance with these standards during actual operations rests with the facility and the NRC.

EPA has for some years conducted special field studies in order to characterize the environmental releases, transport, and impact of radionuclides from fuel cycle facilities.  These have included detailed general studies at pressurized and boiling water reactors, a fuel reprocessing facility, and at mill tailings piles.  In addition, specialized studies of iodine pathways and of nitrogen-16 radiation at reactors have recently been carried out.  These studies will continue in the future.  They are of invaluable assistance in providing soundly based knowledge for assessing the behavior of environmental releases of radioactive materials, and in judging the adequacy of environmental models used for assessing compliance.  The measurement capabilities developed for these studies may also prove useful and will be available

for situations in which the NRC needs assistance in field verification of compliance.

## Timing of Implementation of the Standards

It is proposed that these standards become effective two years from the date of promulgation, with the exception of those for krypton-85 and iodine-129, which are proposed to become effective in 1983.

All existing reactors are now or will shortly be in compliance. In any case, it is considered reasonable to expect that any reactor facilities not now in compliance with Appendix I will be by 1978, three years after its issuance and the earliest possible implementation date for these standards. The question of timing of implementation of the standards is not significant, therefore, as it applies to reactors.

Only one fuel reprocessing facility is now likely to become operable by 1978, and, on the basis of its environmental statement and EPA's assessment of its projected control capabilities, this facility should be able to achieve compliance with the standards at that time. Future compliance with requirements for krypton and iodine releases will depend on the installation of additional controls by 1983. In this regard, it should be noted that the effective date of 1983 for this portion of the standard applies to any release of these nuclides after that date, not to nuclides produced in fuel irradiated after 1983.

Implementation of these standards at milling facilities will in many cases require the installation of updated dust collection equipment, and institution of dust control methods at tailings piles. This equipment is

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commonly available in commerce. The standards do not apply retroactively to offsite windblown tailings, nor to tailings piles at sites no longer licensed. In a few instances large instabilized tailings piles may exist at sites with active licenses. The Agency has these special situations under study.

SUPPLEMENT    B


DOSE-EFFECT ASSUMPTIONS USED AS THE

BASIS OF THE PROPOSED STANDARDS

## DOSE-EFFECT ASSUMPTIONS USED AS THE BASIS OF THE
## PROPOSED ENVIRONMENTAL RADIATION STANDARDS FOR THE URANIUM FUEL CYCLE

Many comments were received concerning the Agency's use of the
linear nonthreshold dose response model for estimating the potential
consequences of doses to populations.  While a few commentors believed
this model was insufficiently protective of public health, the majority
of comments questioned the Agency's health effects estimates in the
belief that they were overly conservative.  These comments were confined
to estimates of cancer risk; the Agency's use of a linear nonthreshold
model to estimate genetic risks, perhaps the largest class of potential
health effects, was not questioned.  The Agency agrees that in certain
cases a linear nonthreshold model could over- or under-estimate somatic
health effects, and has adopted a policy of utilizing other dose-effect
models where clinical data clearly indicate better risk estimates can be
made using other assumptions. For example, the Agency has stated that it
is highly probable that a threshold dose is required for the induction
of skin cancer, and therefore such cancers were excluded from considera-
tion in developing these standards(1).

No specific data was presented by commentors to indicate that any
non-linear dose response model is applicable to exposures from the
uranium fuel cycle. Rather, frequent reference was made to a statement
by the NCRP(2) that extrapolation from the rising portion of dose-

incidence curves derived from data obtained at high doses and dose rates
cannot be expected to provide realistic estimates of the actual risk of
cancer from low level doses of low LET radiation.  The Agency agrees
that dose incidence curves must be interpreted with care, but believes
much human data, such as that discribed in the NAS-BEIR Report (3), is .
useful for estimating radiation risks.

Three factors have been identified by the NCRP as influencing the
validity of interpolation between zero dose and effects and existing
data based on the linear nonthreshold hypothesis: dose, dose rate and
the LET (linear energy transfer) of the radiation in question(2).  For
high LET radiations, such as alpha particle irradiation due to effluents
from the Uranium Fuel Cycle, NCRP seems to accept the use of linear
nonthreshold hypothesis.  In the case of low LET radiation, such as from
effluents emitting beta particles and gamma rays, the Agency accepts the
fact that the epidemological basis for risk estimates is less straight-
forward and indeed discussed the uncertainties in its technical documents
offered in support of these standards(1).  The Agency is aware that for
low LET radiations in-vitro cell killing experiments generally show
reduced effects at low dose rates, indicating that repair of cell-
killing damage may be taking place.  The case for repair of precar-
cinogenic injury, however, is not nearly as clear-cut.  Demonstrations
of decreasing cancer induction at low dose rates have been confined to a
few studies utilizing laboratory animals, most often mice.  These

studies provide conflicting results depending on the species, pattern of irradiation and even the sex of the animals. The decreases in effects observed are in any case relatively small; about a factor of 2-5, but not several orders of magnitude as suggested by some commentors. It is important to note that the effect of dose rate on radiocarcinogensis in animals is not likely to provide an adequate predictor for human risk, since both the life span and the pattern of cancer induction following irradiation are different in man and animals. Nor is it necessary to limit consideration of this question to animal data. There is some cancer incidence data on the effect of dose rate on humans, unfortunately not cited in NCRP 43, which indicates that low dose rates may be equally or more carcinogenic, particularly for protracted exposures(3,4). Until unequivocal contradictory data on radiocarcinogenesis in humans is available that indicates  protracted low dose rate exposures are less carcinogenic than acute exposure at high dose rates, the Agency considers allowance for reduced injury due to low dose rates too specu-lative to be made part of the basis for standards developed to protect public health. While the Agency does not overrule the possiblity that such data may become available in the future, it does not believe sufficient data exists now to warrant a revision in its somatic health effect estimates based on dose rates.

A separable question from dose rate effects is the question of interpolation from high doses to low doses. The point is often made that interpolation from high doses over-estimates risk if made from a

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portion of the dose response curve where the number of cancers is in

proportion to the square of the dose. However, as pointed out in the

Agency's technical documents(1), interpolations from effects observed

following high doses may also under-estimate the number of cancers

induced because cell killing at high doses substantially reduces the

number of cells at risk for radiocarcinogenic transformations.


There is growing evidence, as suggested in NCRP 43(2), that the

Kellerer-Rossi model for initial radiation injury (not radiocarcinogenesis

per se), which predicts a summation of linear and dose squared response,

is useful for interpreting at least some radiation effects data.

However, the available data in support of this model indicate that at

doses less than about 100 rem the linear, not the dose squared, term

dominates the predicted response. Most, but not all, of the health

effect estimates given in the BEIR Report are based on data that include

at least one point for doses less than 100 rems. Therefore, it is not

considered likely that Agency estimates of radiation-induced cancer are

greatly overestimated by the use of BEIR results. In a few cases it is

possible to test for this effect directly by comparing the results of

human experience at high and low doses(4). Such studies show little

difference in effects per rem and may, in fact, indicate an increased

effect at low doses, particularly in cases where the radiation exposure

is protracted over relatively long periods of time. Again the linear

nonthreshold hypothesis cannot be characterized as being overly con-

servative. The Agency recognizes that the interpolation of risk estimates

for humans from high to low doses is uncertain(5), but believes this is a more prudent public health policy than extrapolating laboratory data on short life span animals to man. None of the comments received indicated why the latter procedure would be preferable.

A number of comments were received expressing the view that the Agency had not recognized the NCRP comment cautioning, "...governmental policy-making agencies of the unreasonableness of interpreting or assuming 'upper limit' estimates of carcinogenic risks at low radiation levels, derived by linear extrapolation from data obtained at high doses and dose rates, as actual risks, and of basing unduly restrictive policies on such an interpretation or assumption"(2). The Agency agrees with the NCRP that only reasonable interpolations are warranted, and believes the proposed Uranium Fuel Cycle Standards are both prudent and reasonable. If there is any disagreement it is in the Agency's adoption of the NAS recommendation that linear interpolation be used as a "best" estimate(3) of risk, and not as an estimate on the "upper limit of risk," which seems to be the current philosophy of the NCRP. The Agency has based its health effects estimates on a continuing review of current scientific information, and it believes these estimates represent the most reasonable interpretation of the available data. It will, of course, review new scientific findings as soon as they become available.

Some commentors expressed the view that numerical estimates of radiation-related risks are of little use if they are not compared with

the risk from other environmental pollutants. While the Agency accepts that such comparisons, including a comparison with "natural background radiation," may place the radiation risk from man's activities in a perspective useful to the public, the Agency does not accept such comparisons as the primary basis for establishing radiation protection standards, since at best it could only result in equity between pollutants - not between costs and benefits. Having made an assessment of potential health risks the Agency believes it is more appropriate to select appropriate limits by means of a cost-effectiveness of health risk reduction methodology, rather than via comparative risk assessment.

A number of commentors noted that the reduction of very small risks even further is either not worthwhile or is not cost-effective. The Agency agrees that the risk to an individual from certain radioactive effluents may often be small. However, unless a threshold for radio-carcinogenesis can be demonstrated, the total risk is not necessarily small, but depends on the number of persons exposed. In developing the proposed standards careful consideration was given to the cost-effectiveness of various levels of risk reduction for the entire exposed population, not just for specified individuals. The standards proposed were chosen so as to avoid the imposition of any unreasonable costs for control. It is the Agency's conclusion that all of the costs incurred will be justified by the concomitant reduction of a potential risk to public health.

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(3) The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, Division of Medical Sciences, National Academy of Sciences, National Research Council, November 1972.

(4) Linearity vs. Nonlinearity of Dose Response for Radiation Carcinogenesis, Brown, J. M., review paper presented at the 20th Annual Meeting of the Health Physics Society, July 15, 1975.

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SUPPLEMENT   C


POTENTIAL LIMITATIONS ON MULTIPLE REACTOR SITES

IMPOSED BY THE PROPOSED STANDARDS

POTENTIAL LIMITATIONS ON MULTIPLE REACTOR SITES
IMPOSED BY STANDARDS FOR THE URANIUM FUEL CYCLE

## Introduction

The number of reactors at a given site could be limited, at least in

principle, by an ambient environmental radiation standard applying to all

activities in the uranium fuel cycle (1,2). In order to examine this

possibility, conclusions developed during the AEC's (now NRC) rulemaking

on as low "as practicable" (ALAP) reactor effluents, AEC and NRC

dosimetric estimates for real sites in environmental statements, the

results of EPA field studies, operating data for reactors, and some

analyses of hypothetical configurations are each examined in turn below.

First, however, we digress for a brief assessment of the number and sizes

of multiple reactor sites to be expected, based on actual commitments by

utilities during the next decade.

## Multiple Reactor Site Projections

Originally, nuclear power reactors were constructed as individual

units, each on its own site. As nuclear power became more attractive

economically and technologically, multiple reactors were ordered for

single sites. A recent listing of all reactors in operation, under construction, or on order (3) reveals that there are only six sites for which as many as four reactor units are presently committed. These four-unit sites are:

| Site | Location | Commercial Operation Expected for Last Unit |
|------|----------|---------------------------------------------|
| Alan R. Barton | Verbena, Ala. | 1987 |
| Hartsville | Hartsville, Tenn. | 1982 |
| North Anna | Mineral, Va. | 1981 |
| Shearon Harris | Newhill, N.C. | 1990 |
| Surry | Gravel Neck, Va. | 1984 |
| WPPSS | Richland, Wash. | 1982 |

TVA also has plans for four more reactor units at as yet unspecified locations, which may or may not be built on the same site. Thus, it is likely to be at least five years before a four unit site could be in operation. No sites containing more than four reactor units are presently committed. Considering the lead time of eight years necessary (from contract award to commercial operation) for a single reactor unit, it will apparently be at least a decade before a five or six unit site could become operational.

## Considerations from the ALAP Rulemaking

One of the basic questions considered by the NRC in the rulemaking for as low "as practicable" discharges from light-water-cooled nuclear power reactor effluents was whether the design objectives of Appendix I to 10 CFR 50 should apply to each reactor or each site. The original

proposal would have applied the basic dose limits to entire sites.

However, in the words of the Commission (4),

> "We have chosen to express the design objectives on
> a per light-water-cooled nuclear power reactor basis
> rather than on a site basis, as was originally
> proposed. While no site limits are being adopted,
> it is expected that the dose commitment from multi
> light-water-cooled reactor sites should be less than
> the product of the number of reactors proposed for a
> site and the per-reactor design-objective guides
> because there are economies of scale due to the use
> of common radwaste systems for multi-reactor sites
> which are capable of reducing exposures."

Later, in a more detailed discussion of this question (4), the Commission

expressed the view:

> "We are also of the opinion that it will be at least
> several years before sites containing as many as
> five light-water-cooled nuclear power plants are
> developed. Consequently, we see no way that design-
> objective guides set on a per-reactor basis can, in
> the near future, result in individual exposures that
> are more than 5% of present-day (10 CFR 20)
> radiation standards. Indeed, we believe that, with
> the required inclusion of all radwaste augments
> justified on a cost-benefit basis and with the
> realization that several reactors cannot physically
> be placed so as to all be a minimum distance from
> the maximally exposed individual, the actual doses
> received by individuals will be appreciably less
> than this small percentage."

Thus, it was the opinion of the Commission that the radiation doses from

multi-reactor sites, containing up to five light-water-cooled nuclear

power reactors, will remain at small percentages of present-day (10 CFR

20) radiation standards, specifically, at less than 25 mrem/yr to the

whole body and 75 mrem/yr to the thyroid.

## Results of NEPA Reviews

For the last few years, the AEC and NRC have filed environmental statements under the provisions of the National Environmental Policy Act; these environmental statements assess the expected performance characteristics for projected nuclear facilities, including nuclear power reactors. Table 1 summarizes the results of these analyses for radioactive releases from all sites projected to contain three or more reactors. The table shows that:

(1) For the eleven such sites analyzed, in only one case is a whole body dose by any pathway greater than 2 mrem/yr projected. The exception, 12 mrem/yr to a hypothetical individual consuming 18 kilograms per year of shellfish collected from the reactor discharge canal, is based upon the assumption that public access to that canal is permitted.

(2) For no site is a maximum dose of more than about 15 mrem/yr to the thyroid of an infant at the nearest farm necessary if reasonable and readily available control measures are instituted.

It must be emphasized that the estimated doses in Table 1 have been calculated using conservative models. Even though the most recent environmental statements employ models specified by regulatory guides which are more realistic than those used in the past, these models are still conservative. Again, in the opinion of the Nuclear Regulatory Commission on Appendix I to 10 CFR 50 (4):

> "It must be understood in discussing the matters of
> calculational conservatism and realism that Appendix
> I means, implicitly, that any facility that conforms
> to the numerical and other conditions thereof is
> acceptable without further question with respect to

## Table 1. Environmental Impacts of Three and Four-Unit Sites

| Site | EIS(Date) [b] | Dose Equivalent Rate (mrem/yr) | | | Thyroid [a] |
| | | Whole Body | | | |
| | | Gaseous | Liquid | Site Gamma [c] | Iodine |
|---|---|---|---|---|---|
| **Four Unit Sites** | | | | | |
| Hartsville | 6/75 | <1 | <1 | <1 | 16[d] |
| Alan R. Barton | 4/75 | <1 | 1.3 | <1 | 2.2 |
| WPPSS(Hanford) [e] | 3/75 | 1.2 | 2.5 | <1 | 9 |
| Surry | 6/72,5/74 | <1 | 2.5 | <1 | 49[f] |
| Shearon Harris | 3/74 | <1 | 12[g] | <1 | 15[h] |
| Vogtle | 3/74 | <1 | <1 | <1 | 4 |
| North Anna | 4/73 | <1 | 1.3 | <1 | 4.9 |
| **Three Unit Sites** | | | | | |
| Davis-Besse | 2/75 | <1 | 2.2 | <1 | 1.8 |
| Pilgrim | 6/74 | 1.3 | <1 | <1 | 2.9 |
| Millstone | 2/74 | <1 | <1 | <1 | 10 |
| Dresden | 11/73 | <1 [i] | <1 | <1 | 7 |
| Indian Pt. | 10/73 | 2.4 | <1 | <1 | <1 |
| San Onofre | 3/73 | <1 | <1 | <1 | <1 |
| Browns Ferry | 9/72 | 2 | <1 | <1 | <1 |
| Oconee | 3/72 | 1 | <1 | NR | 5 |

FOOTNOTES

(a) Dose equivalent to infant thyroid via cow-milk-pathway at nearest farm
(b) All are final environmental statements except Barton, Davis-Besse, and Pilgrim, which are draft statements.
(c) 500 hours unshielded occupancy of boundary per year
(d) The applicant's facility design, as proposed, would result in a dose of 74 mrem/yr, which was not deemed "as low as practicable" by the NRC staff. Addition of a turbine building ventilation treatment system could reduce the total dose to about 16 mrem/yr, as indicated in the statement.
(e) Does not include dose equivalents from the Hanford N-Reactor, which is a light-water-cooled, graphite moderated reactor.
(f) 98% of the release is from the condenser air ejector and steam generator blowdown tank vent pathways of Units 1 and 2 and can be eliminated or drastically reduced through simple modifications of existing control equipment.
(g) Assumes public access to cooling water discharge canal and annual consumption of 18 kg of fish and mollusks raised in discharge.
(h) Monitoring and appropriate operational practices will be required by NRC to assure that dose levels do not exceed 15 mrem/yr; NRC considers the calculated dose without such measures (28 mrem/yr) very conservative (i.e., the actucal dose will be lower).
(i) Based on augmented system committed by applicant (p.11-40 of EIS)

NR - Not Reported

section 50.34a...The numerical guidelines are, in
this sense, a conservative set of requirements and
are indeed based upon conservative evaluations."

In any event, the results presented in Table 1 indicate that for all

multi-reactor sites for which environmental assessments are available,

the maximum projected dose is less than 5 mrem to the whole body, even.

under the highly unlikely presumption that the maximum whole body doses

for gaseous and liquid effluents add arithmetically. Thyroid doses would

limit the number of such reactors at a given site to no greater extent

than do whole body doses. This conclusion is, of course, in harmony with

that reached by the NRC that sites containing as many as five light-

water-cooled nuclear power reactors would result in individual exposures

that are appreciably less than 25 mrem/yr to the whole body and 75

mrem/yr to the thyroid.

## Results from Field Studies

In addition to the estimates of dosimetric impact made using

"realistically conservative" calculational models, the EPA and its

predecessor organizations have conducted detailed surveillance programs

at selected facilities (5-9). These studies have confirmed the accuracy

of reported effluents of noble gases and liquids, but appear to reveal

significantly lower iodine concentrations in milk than projected by

models for the milk pathway currently used for environmental analysis.

Field studies conducted by the EPA at Dresden (Unit 1), Yankee Rowe,

and Haddam Neck (formerly Connecticut Yankee) have shown the following

maximum individual doses to the various organs listed (5-8):

## Maximum Individual Dose (mrem/yr)

| Organ | Dresden | Yankee | Haddam Neck |
|---|---|---|---|
| Whole body | 8.0 | 3.0 | 3.8 |
| Thyroid | 0.74 | 0.006 | 6.0 |
| Bone | 0.026 | 0.20 | 3.0 |
| GI (LLI) | 0.008 | 0.26 | 0.4 |

It should be noted that these values are absolute maximum doses for each

organ; all pathways possibly contributing dose to a particular organ were

summed to arrive at the above totals. These doses thus presume that an

individual could be simultaneously exposed to all pathways of exposure

and that he would receive the maximum possible dose from each pathway.

Thus, these doses are extremely unlikely to have been received by any

real individuals, as was pointed out by the authors of the Dresden and

Yankee studies (8):

> "...a farmer near Dresden may eat beef, green
> vegetables, and drink milk, but he would not also
> eat 100 gms of fish per day that had been caught at
> Starved Rock Dam, neither would he consume Peoria
> drinking water, nor does he reside in the areas for
> which inhalation and external whole-body exposures
> were calculated. Consequently, actual radiation
> exposures to existing populations in the vicinity of
> both nuclear power plants are less than the total
> dose rates listed..."

Furthermore, most of the whole body dose listed for the pressurized water

reactors (PWRs), Yankee Rowe and Haddam Neck, result from direct

radiation originating from stored radioactive waste (gaseous and liquid

storage tanks). This exposure may be minimized by simple shielding or

careful placement of these tanks relative to the site boundary.
Virtually all of the thyroid dose and bone dose at Haddam Neck results
from the hypothetical consumption of fish (18 kilograms per year) caught
in the discharge canal. Almost all of the whole body dose listed for
Dresden results from exposure to the gaseous effluent (principally noble
gases) discharged from the stack; boiling water reactors (BWRs) such as
Dresden are presently augmenting (or have already augmented) their noble
gas treatment systems to provide additional dose reduction factors of 8
to 180 beyond those in force at the time the above studies were carried
out (2). The three reactors studied are also of early design. Reactors
going into operation today or in design and construction stages
incorporate considerably more sophisticated radwaste treatment systems
having larger processing capacities, greater cleanup efficiency, and
increased flexibility.

Doses due to gamma radiation (directed and scattered, or "shine")
originating onsite can be significant at BWR sites because of the
circulation of activation-produced nitrogen-16 through the turbines and
associated equipment, particularly the moisture separators. The EPA
field studies discussed above considered the whole body dose from direct
gamma radiation only for the PWR field studies (Yankee Rowe and Haddam
Neck). Field measurements made by the EPA, ERDA, NRC, and others have
shown that dose rates on the order of 10 mrem/yr (whole body) at 500
meters are possible without supplementary shielding of turbine building
components; these dose rates, however, decrease very rapidly with
distance so as to produce very small population doses (10-13). In

addition, dose rates are very dependent upon the design and layout of the turbine and its associated equipment. Appropriate design of shielding and location of turbine components relative to the site boundary can assure that offsite doses from "turbine shine" are minimized. The siting of many reactor units at a single site should also result in significantly smaller offsite doses from turbine "shine," as the exclusion distance increases with the number of reactor units on a site. According to a recent study (14), the exclusion distance averaged 460 meters for single unit BWRs and 860 meters for twin-unit BWR sites; for PWRs, single units sites averaged 750 meters, while twin-unit sites averaged 900 meters. Since the dose from turbine "shine" falls off very rapidly with distance, such doses should be significantly reduced for multi-reactor sites. For example, using the data from the most recent study (13), the dose rate falls off by a factor of five as the distance increases from 460 meters to 860 meters. Therefore, it is to be expected that dose rates from turbine "shine" at multi-reactor sites will not be significant compared to those from the single unit sites at which field studies have taken place.

Studies of iodine pathways and potential thyroid doses have been conducted jointly by EPA and AEC over the past two years at the Dresden, Monticello, Oyster Creek, and Quad Cities sites (9). The available results present a consistent picture of iodine concentrations in milk less than these projected by models for the milk pathway currently used for environmental analyses.

## Results from Reactor Operation

In addition to conservative environmental dose pathway models, radionuclide source term models have generally been conservative. For example, fuel experience for PWRs has been much better than the 0.25% fuel leakage rate now used as a design basis for calculating environmental releases. Westinghouse, which has manufactured the great majority of operating PWRs, reports that fuel integrity has generally been in the neighborhood of 99.98% (i.e., a fuel leakage rate of 0.02%) for zircaloy-clad fuel. Exceptions to this high level of fuel integrity occurred in 1969-1970, when hydriding lowered fuel integrity to the 99.8-99.9% range, and in 1972, when fuel densification lowered fuel integrity to the 99.9% range (15). On the other hand BWRs, which have typically been designed for fuel leakage corresponding to the release of 100,000 uCi/sec of noble gases from the air ejector, after a nominal 30 minute delay, exhibit a more variable performance. Figure 1 shows that this design value had yet to be reached by BWRs operating through 1973; indeed, most were very much below the design value (16). Recent data, however, indicate a rising trend of releases from BWRs, and EPA is maintaining a continuing surveillance of this trend, which may indicate that the present design basis is too low to provide adequate assurance that Appendix I design objectives will be satisfied in actual operation. In general, however, fuel integrity at PWRs and for pre-1974 BWR performance has been considerably better than predicted by conventional source term models used in environmental analyses.
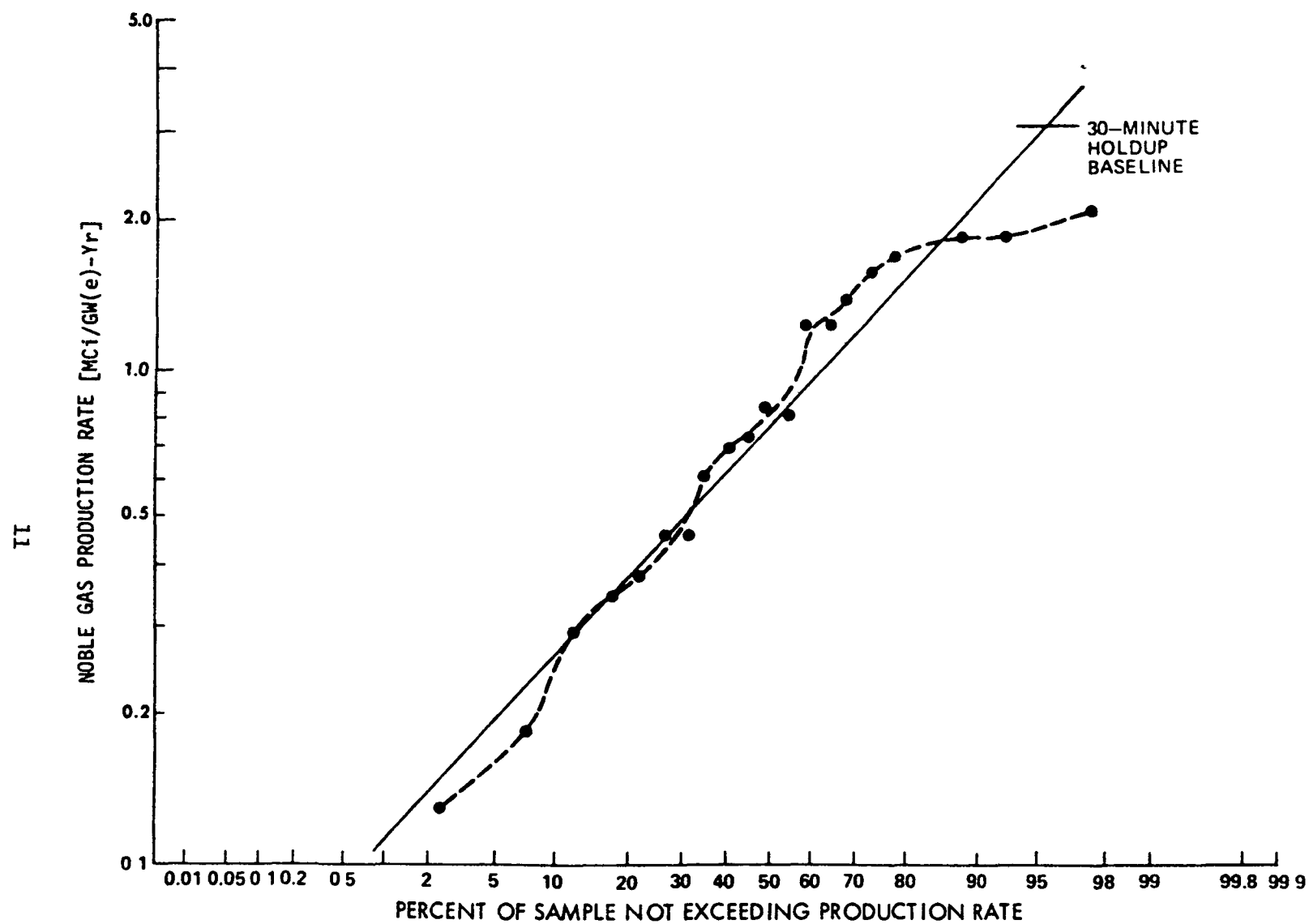
Figure 1 Distribution of noble gas releases in 1971-73 for boiling water reactors that commenced operation after 1968. The solid line is a fitted log normal distribution. (Ref. 15)

A second important consideration with respect to conservatism in source term models is the fact that, especially for PWRs, effluents are postulated for inplant pathways which require simultaneous levels of degradation of several parameters in order to lead to a postulated release to the environment. For example, effluents from the PWR secondary system (e.g., steam generator blowdown vent or condenser air-ejector exhaust) require the simultaneous existence of a "design basis" assumed fuel leakage and a "design basis" assumed steam generator leakage rate of primary coolant into the secondary coolant. Since the probability of each "standard" assumption is generally significantly less than one, the probability of both occurring at the same time must be smaller than either of the individual probabilities. Thus, if the annual probability of having the "design basis" number of fuel failures is five percent and the probability of having a "design basis" primary to secondary leak is twenty percent, the probability of operating a PWR with "design basis" fuel leakage and primary to secondary leakage is of the order of one percent. In spite of this, light-water-cooled reactors have been evaluated as if these "design basis" conditions occur simultaneously, for periods of time comparable to a year (17).


## Analysis of the Additivity of Doses from Multiple Facilities

Similar considerations apply to the assessment of doses from multiple facilities on a single site. A variety of site specific factors exist, including the site size, the relative location of individual facilities on the site, and economies available through utilization of

design incorporating shared control measures, each of which mitigate

against arithmetic additivity of doses to a maximum exposed individual

outside the site boundary. In general, these effects are quite

significant, as is reflected by the low doses projected for those sites

which have been subjected to analysis, as, for example, in the

environmental statements quoted above. Indeed, these sites project lower

doses than many single unit sites. In addition, however, there is

significant operational flexibility available at a multi-unit site not

available to sites containing single or double units. For example, if a

reactor at a four-unit site is experiencing a severe rate of fuel

failure, the output of the site could be maintained at a respectable 75%

of capacity while that reactor is serviced, by operating the remaining

units at full fuel capacity, a degree of flexibility not available to a

one- or two-unit site without calling upon another portion of the power

grid to take up the loss of capacity.

In addition to the above considerations, which in actual situations

should generally be overriding, it is, however, also instructive to

consider the following hypothetical example. Assume that all units on a

site are located at exactly the same point, and that each is designed to

no more than conform exactly, using "design basis" assumptions, to the

design objective doses specified by Appendix I (say, 5 mrem whole body

dose via the air pathway) to some common hypothetical worst case

receptor. Assume further, since under Appendix I this dose is to be

exceeded only in "temporary" and "unusual" situations (4), that one may

assign some reasonable probability that, on an annual basis, the design

objective dose for any single unit will not, in fact, be exceeded. For example, the 0.25% fuel failure assumption currently used as a design basis for PWRs is not exceeded, on the basis of current operating, history, at least 95% of the time. What then, is the dose that can be expected to be not exceeded at the same confidence level (95%) for 4, 5, or 6 such units? That the answer is not 4, 5, or 6 times 5 mrem/yr is obvious. The exact result is dependent upon the variance of the operating data, and, to a lesser degree, the shape of the distribution of the data. A statistical analysis utilizing actual operating data for PWRs and BWRs yields the following projections (18):

Dose Levels (mrem/yr) that will be Satisfied 95% of the Time†

|  | 4 Units | 5 Units | 6 Units |
| --- | --- | --- | --- |
| PWR | 15 | 18 | 21 |
| BWR | 16 | 19 | 22 |

†For single units which each satisfy Appendix I at the 95% confidence level; each value has an uncertainty of approximately 1 mrem/yr.

Each of these values is significantly lower than that predicted by an assumption of additivity, even for the extreme case of colocation of all units, no exercise of operational flexibility, and design for the maximum release permitted by Appendix I considered here.


## Conclusion

On the basis of a) results projected by the AEC and NRC for all multi-unit sites presently committed, b) the flexibility available through proper selection and utilization of future sites, c) the

conservative nature of design dose calculations, as opposed to the applicability of these standards to exposures actually received, d) the nonadditivity of design basis dose contributions from single units, and e) the operational flexibility available to sites with multiple units, it is concluded that the proposed standards can be readily achieved at all presently planned and all properly designed future multi-unit sites of up to at least five units. It is further noted that in "unusual" circumstances during which the design objectives specified for light-water-cooled reactors by Appendix I may be "temporarily" exceeded (4), that the variance provision of the proposed standards would permit continued operation in times of necessity. Questions associated with even larger configurations of units, such as nuclear energy centers, are addressed separately.

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SUPPLEMENT    D


AN ANALYSIS OF CONTROL OPTIONS FOR NITROGEN-16

OFF-SITE SKYSHINE DOSES AT BOILING WATER REACTORS

# AN ANALYSIS OF CONTROL OPTIONS FOR N-16 OFFSITE SKYSHINE DOSES AT BOILING WATER REACTORS

## Introduction

The turbine system at a boiling water reactor (BWR) is a potentially significant source of radiation due to the presence of nitrogen-16, a relatively short-lived ($t_{\frac{1}{2}}$=7.14 sec), high energy (2.75 Mev (1%), 6.13 MeV (69%), and 7.11 MeV (4.9%)) gamma emitter in the steam leaving the reactor. Nitrogen-16 is produced in the reactor core by neutron activation of oxygen in water, and, although short-lived, can be present in the turbine system in significant quantities due to the rapid transit of steam from the reactor vessel through the turbine system and to the condenser. The result is a flux of direct and scattered gammas which can result in high occupational exposure rates in and close to the turbine building, as well as potentially significant exposure rates to members of the public beyond site boundaries near the turbine building.

## Sources

Detailed expositions of nitrogen-16 sources are presented in the safety analysis report for the General Electric standard boiling water reactor, the BWR/6 (1), and for operating BWR's in a comprehensive report recently released by General Electric (2). In these reports a nitrogen-16 activity concentration of 50 μCi/gm of steam at the reactor nozzles is assumed, based on experimental measurements of contact dose rates on cross-around pipe sections of operating BWRs. Other analyses (3,4) have assumed nitrogen-16 activities of up to 100 μCi/gm of steam at the

nozzles; however, this is probably due to the desire for conservatism in the design of shielding.

In a typical modern boiling water reactor, steam flows directly from the reactor nozzles through the main steam header to the high pressure turbine (HPT). Steam extraction is also made from this flow path for steam to the steam jet air ejector (SJAE), feed water heaters (FWH), gland seal system, and the moisture separater/reheater units (MSRH). Steam leaving the HPT is routed through the shell side of the MSRH's, where it is dewatered and reheated for injection into the low pressure turbines (LPT). Steam extractions are also made at the HPT, MSHR's, and in several places along the LPT for the various feedwater heater stages (usually 6).

Typical delay times to and transit times through these components are shown in Table 1. At a concentration of 50 µCi/gm of steam, the nitrogen-16 source term at the nozzles is 100 Ci/sec. Thus, it is obvious that the potential exists for considerable equilibrium activity to be present in these turbine system components.

Table 2 lists the calculated inventories for the various turbine building components. The dose significance of these sources depends on the shielding (both exterior and self-shielding of components) as well as the geometry of the component layout. The typical order of the dose significance by component is a) moisture separater/reheaters, b) intermediate piping, c) high pressure turbine, and d) all other components.

## Turbine Building Configurations

The configuration in which components are placed in a turbine

building has undergone several changes in recent years. Several

different turbine manufacturers have supplied turbines for BWR reactor

plants and component layout has varied as a function of both turbine

manufacturer and of architect-engineer. Turbines have been supplied by

General Electric, Westinghouse, and Kraftwerk-Union, for example, and

facilities using BWR's have been engineered by a variety of architect-

engineering firms. The major significant system design changes have been

with respect to the placement of moisture separaters and reheaters.

Earlier BWR designs had vertically-oriented moisture separaters and

separate reheaters located on the mezzanine level of the turbine building

(below the operating floor) as shown in Figure 1. Considerable

shielding was afforded by the concrete structure of the turbine building

around these components, and, particularly above, by the operating floor.

For a variety of engineering reasons, including increased efficiency

of turbine operation, reduction in building size, and reduction in time

of construction, recent designs have incorporated horizontally-oriented

combined moisture separaters and reheaters located above the turbine

building operating floor level, as shown in Figure 2. The high

equilibrium nitrogen-16 activity levels in tube and shell side of these

systems, combined with the relative lack of self-shielding, compared to

that of the thick steel shells and massive internals of turbines, result

in these "exposed" MSRH's and their supply and return piping producing a

potentially high gamma flux in comparison with all other components.

A system which can perhaps be considered an example of a "worst case" is the combination of a General Electric BWR with a Westinghouse turbine system. In this case the steam piping runs overhead from the top of the HPT to the top or side of the MSRH. Since there is considerable nitrogen-16 activity in these pipes, they can provide a significant additional source of gamma exposure beyond the MSRH's themselves.

## Dose Assessment

The gamma flux existing at a point outside a turbine building due to sources of nitrogen-16 inside is difficult to calculate. Gammas may arrive at a given point by direct paths, by scattering in shielding and other components, or from air scattering, as shown in Figure 3. The shielding geometry is complicated due to the variety of component shapes and locations, and each component also has different self-shielding factors for the gammas involved.

A variety of types of computer codes have been developed to calculate the air-scattered contrubution to the gamma exposure field (see, for example, refs. 2,6,7). The potentially most accurate of these are Monte Carlo transport codes. However, these models have not been verified by EPA, and they are sufficiently complex and expensive to prohibit performing such analyses on a case-by-case basis. No discussion of analytical techniques for quantitatively analyzing these exposure rates based on transport codes was undertaken, although the results of some calculations performed by industry (5) provide the basis for the present comparison of several options.

Insight into the relation between various shielding options and anticipated dose rates can be obtained, however, through an examination of existing shielding studies in conjunction with field measurement studies. This examination indicates the principal contributors to and magnitudes of potential doses and permits an informed, if not detailed, understanding of what might be required to reduce such doses.

## Shielding of Components

Because of the high radiation field resulting from nitrogen-16 activity, existing turbine systems are already well-shielded. This is not primarily because of consideration of doses beyond site boundaries, but due to the need to comply with existing occupational exposure limits. In order to restrict the extent of high radiation areas adjacent to turbines and to allow more frequent or even uncontrolled access to other areas in the turbine building, the turbines and MSRH's are heavily shielded. Usually this shielding consists of a thick concrete "shadow shield" surrounding the turbine (as much as 4 ft thick), and upward extension of the turbine building lower side walls (up to 3 ft thick) to shadow-shield the MSRH's. While such shielding substantially reduces the direct components of the gamma flux, air-scattered contributions from gammas leaving the unshielded top of the turbines and MSRH's can still produce considerable exposure rates. Therefore, often as a design option, many recent designs have included concrete shields (up to 20" thick) over the MSRH's and vertical steel plating running between the turbines and MSRH's to reduce this air-scattered flux (see Figs. 4,5). In order to assess the effectiveness of such additional shielding as a

means to reduce site boundary doses we have chosen to analyze a variety

of such shielding options for the turbine building component

configuration shown in Figure 4. The assumption is made that concrete

walls are already in place around the MSRH/turbine area as shown to allow

required access in the remainder of the turbine building area within

applicable limits for occupational exposure. These walls are assumed to

consist of three feet of reinforced concrete; this thickness will provide

an attenuation of approximately 99.7% of the incident gamma flux

(neglecting buildup), leaving only the scattered flux as a potentially

significant contributor to the off-site dose.

Such a characterization of skyshine as the principle source of

exposure from nitrogen-16 at distances greater than a few hundred meters

from the turbine building is supported by a recent field study performed

at the Cooper Nuclear Station by EPA and ERDA (8). Cooper station is a

BWR with a Westinghouse turbine and horizontally-oriented moisture

separaters located on the turbine building operating floor. Field

measurements were made by EPA in February, 1975, and by ERDA's Health and

Safety Laboratory in April, 1975. Cooper is a reasonable example of the

"base" case turbine building discussed above, since shielding consists of

side walls only, although in this case these consist of 3 ft of high

density concrete. A significant finding of the study was that nearly

100% of the dose measured was due to air-scattered (skyshine) gammas.

The contribution to dose of the direct flux was negligible.

Referring to Table 3, it can be seen that for the base case the

total net equivalent activity above the turbine operating floor is 34 Ci.

Out of this total, 21 Ci are associated with the moisture
separater/reheater and 10.3 Ci are associated with the intermediate
piping.

The shielding options considered, calculated doses, and anticipated
costs are presented in Table 4. These have been derived in part from
information provided the Agency by General Electric (5). With these
options and their associated dose rates as a basis, and using Means 1975
Building Construction Cost Data (9), we have made independent cost
estimates for installing the additional shielding required by each of the
options considered. The costs presented do not include any additional
basic building structure which might be required within the turbine
building to support the additional weight of the shielding, because for
most of the cases considered the additional weight involved does not
appear to require any additional support beyond that already available in
the basic structure supporting the turbine and other components. The
costs presented here are appropriate to plants in the design stage, and
would not necessarily apply to retrofit situations.

All cases above the base case include the cost of poured-in-place
reinforced concrete, which is supported by an assembly of steel girders
bridging the MSRH's between the exterior turbine shielding wall and
inside panel wall. The inside panel includes steel columns to provide
additional support for the overhead assembly. The dimensions required
for each of two overhead shields are conservatively estimated to be 140'
long by 35' wide. The inside panel walls are assumed to be 140' long by
25' high. The concrete for exterior side walls and end walls is assumed

to be already present as the "base case." Costs of materials, installation, engineering, financing, overhead, and profit, were based on standard estimating assumptions (10). Details of the estimation procedure used are available upon request. Table 4 provide a summary of costs for the various shield options, and Figure 6 displays annual dose at 500 meters vs. cost of shielding.

Doses are presented for the various shielding options both as calculated by the industry and as projected from values measured in the field. The field study resulted in data which indicates that the calculated doses are high by approximately a factor of two. In addition, the assumption of 100% occupancy, no additional shielding by offsite building structures, and annual operation at 100% power are considered to be unreasonably conservative assumptions for estimating real doses to individuals at real sites. It is concluded, therefore, that it should be readily possible to restrict the dose from nitrogen-16 skyshine to a real individual located at reasonable distances from the center of the turbine building for realistic occupancy times to less than 2 mrem/yr. These dose levels should be attainable for no more than approximately $250,000 and even these costs should be incurred only in those few instances where actual site boundaries are so close to turbine buildings as to create the possibility of significant offsite exposures from nitrogen-16 sources.

## Notes

1. BWR/6 Standard Safety Analysis Report, General Electric Company, NEDO 10741, Vol. 8.

2. Rogers, D. R., "BWR Turbine Equipment Nitrogen-16 Radiation Shielding Studies," General Electric Report NEDO-20206 (December 1973).

3. "Radiation Shielding Design and Analysis Report - Nine Mile Point Nuclear Station Unit 2," Stone & Webster Engineering Co., RP-6, (January 1974).

4. Preliminary Safety Analysis Report - Newbold Island Nuclear Station, Public Service Electric and Gas Company of N.J., February 1970.

5. Information provided EPA by General Electric and Bechtel Engineering Staff, (January 1975).

6. Woolsen, W.A., A.E. Profio, D.L. Huffman, "Calculation of the Dose at Site Boundaries from Nitrogen-16 Radiation in Plant Components," JRB 72-507 LJ, JRB Associates, (Dec. 1972).

7. Ward, J.T., Jr., "A Dose Rate Kernel for Air-Scattered Nitrogen-16 Decay Gamma Rays," Ph.D. Thesis, University of California, Berkley

8. Phillips, C., Lowder, W., Nelson, C., Windham, and Partridge, J., Nitrogen-16 Skyshine Survey at a 2400 MW(t) Power Plant, EPA -520/5-75-018.

9. Godfrey, R.G., Editor, "Building Construction Cost Date 1975," 33rd Ed., 1974, Robert Snow Means Company, Inc.

10. The following markups were applied to materials and installations: 25% overhead and profit, 2.5% engineering, 10% contingency. A short term financing factor of 1.375 was then applied to the total, representing a 10%/per annum financing cost over a period of three years.

TABLE 1.

$N^{16}$ CHARACTERISTICS OF A STANDARD BWR TURBINE SYSTEM [1]

| Component | Decay Time at Inlet (seconds) | Estimated Mass Inventory (lbs) | Mass Flowrate ($lb/hr \times 10^{-6}$) | Component Transit Time (seconds) |
|---|---|---|---|---|
| Main Steam Line and Header System | | | | |
| a.  Reactor Nozzle to Main Steam Header | 0.00 | $8.933 \times 10^3$ | 15.396 | .2.09 |
| b.  Main Stream Header to HPT | 2.09 | $\underline{4.464 \times 10^3}$ | 14.764 | 1.09 |
| | | $13.397 \times 10^3$ | | |
| High Pressure Turbine | 3.18 | $3.784 \times 10^2$ | 14.748 | 0.0924 |
| Low Pressure Turbines | 5.86 | $7.611 \times 10^2$ | 10.678 | 0.257 |
| Moisture Separator Shell-Side (Steam) | | | | |
| a.  Inlet to Vanes | 4.29 | $1.256 \times 10^3$ | 13.171 | 0.343 |
| b.  Vanes | 4.64 | $3.00 \times 10^2$ | 11.460 | 0.0942 |
| c.  Vanes to Outlet | 4.73 | $\underline{2.119 \times 10^3}$ | 10.904 | 0.700 |
| | | $3.675 \times 10^3$ | | |
| Moisture Separator Shell-Side (Liquid) (Vanes, Drain Trough) | 4.64 | $4.059 \times 10^3$ | 1.712 | 8.54 |

TABLE 1 (Continued)

| Component | Decay Time at Inlet (seconds) | Estimated Mass Inventory (lbs) | Mass Flowrate (lb/hr x $10^{-6}$) | Component Transit Time (seconds) |
|---|---|---|---|---|
| Moisture Separator Drain System | | | | |
| a. Steam | 4.73 | | 0.5554 | |
| b. Liquid | 13.18 | | 1.712 | |
| First Stage Reheat System | | | | |
| a. Supply Pipe - HPT to Tube Inlet | 3.27 | $2.058 \times 10^2$ | 0.7011 | 1.06 |
| b. Tubes | 4.33 | $\underline{6.424 \times 10^3}$ | 0.7011 | 33.0 |
| | | $6.630 \times 10^3$ | | |
| Second Stage Reheat System | | | | |
| a. Supply Pipe—Main Header to Tube Inlet | 2.09 | $2.80 \times 10^2$ | 0.6145 | 1.64 |
| b. Tubes | 3.73 | $\underline{5.811 \times 10^3}$ | 0.6145 | 34.0 |
| | | $6.091 \times 10^3$ | | |
| First Stage Reheat Drain System | 37.3 | | 0.7011 | |
| Second Stage Reheat Drain System | 37.8 | | 0.6145 | |

TABLE 1 (Continued)

| Component | Decay Time at Inlet (seconds) | Estimated Mass Inventory (lbs) | Mass Flowrate (lb/hr x $10^{-6}$) | Component Transit Time (seconds) |
|---|---|---|---|---|
| Piping System - HPT to MS/RHR | 3.27 | $3.717 \times 10^3$ | 13.171 | 1.02 |
| Piping System - MS/RHR to LPT | | | | |
|   a.  MS/RHR to CIV | 5.43 | $6.857 \times 10^2$ | 10.904 | 0.227 |
|   b.  CIV | 5.66 | $2.852 \times 10^2$ | 10.678 | 0.0962 |
|   c.  CIV to LPT | 5.75 | $2.812 \times 10^2$ | 10.678 | 0.0948 |
| | | $1.252 \times 10^3$ | | |
| First Stage FWH and Extraction System | | | | |
|   a.  Extraction Point 4 | 6.12 | | 0.1016 | |
|   b.  Extraction Point 5 | 6.12 | | 0.6017 | |
| Second Stage FWH and Extraction System | 6.12 | | 0.6301 | |
| Third Stage FWH and Extraction System | 6.12 | | 0.7344 | |
| Fourth Stage FWH and Extraction System | 6.12 | | 0.4016 | |
| Fifth Stage FWH and Extraction System (Excluding MS Drain System) | 3.18 | | 0.0126 | |

TABLE 1 (Continued)

| Component | Decay Time at Inlet (seconds) | Estimated Mass Inventory (lbs) | Mass Flowrate (lb/hr x $10^{-6}$) | Component Transit Time (seconds) |
|---|---|---|---|---|
| Sixth Stage FWH and Extraction System (Excluding Reheater Drain Systems) | 3.27 | | 0.857 | |
| Condenser (Excluding return from FW Turbine) | 6.12 | | 8.207 0.0016 | ˜30 (liquid) ˜ 1 (gas) |
| Hotwell (Excluding return from FW Heaters, etc.) | ˜36 | | 8.207 | |
| SJAE First Stage System | | | | |
|    a. Off-Gas | ˜7 | | 0.0016 | |
|    b. Driving Steam Supply Line | 2.09 | $1.12 \times 10^{1}$ | 0.0180 | 2.24 |
|    c. First Stage Driving Steam | 4.33 | | 0.0080 | |
| Recombiner System (Second Stage Air Ejector Driving Steam) | 4.33 | | 0.0100 | |
| Gland Seal System | | | | |
|    a. From HPT | 3.27 | | 0.0186 | |
|    b. From Valve Stem | 3.18 | | 0.0029 | |
| Feedwater Turbine System | 5.66 | | 0.2259 | |

## Table 2.
### $N^{16}$ Inventories For A Standard BWR Turbine System [1]

| Component | N-16 Inventory (Curies) |
|---|---|
| Main Steam Line and Header System | 263 |
| High Pressure Turbine | 6.3 |
| Low Pressure Turbines (1) | 9.8 |
| Moisture Separator and Reheater Shell-side Steam | 53 |
| Moisture Separator Shell-side Liquid | 41 |
| Moisture Separator Drain System | 56 |
| First Stage Reheat System (2) | 33 |
| Second Stage Reheat System (2) | 32 |
| First Stage Reheat Drain System (3) | 1.4 |
| Second Stage Reheat Drain System (3) | 1.1 |
| Intermediate Piping System - HPT to MS/RH | 59 |
| Intermediate Piping System - MS/RH to LPT | 17 |
| First Stage - FWH & Extraction System (4) | 26 |
| Second Stage - FWH & Extraction System (4) | 23 |
| Third Stage - FWH & Extraction System (4) | 27 |
| Fourth Stage - FWH & Extraction System (4) | 15 |
| Fifth Stage - FWH & Extraction System (Excluding Moisture Separator Drain System Activity Listed Above). | .6 |
| Sixth Stage - FWH & Extraction System (Excluding First and Second Stage Reheat Drain System Activities Listed Above) | 42 |

Table 2 (Continued)

| Component | N-16 Inventory (Curies) |
|---|---|
| Condenser (Excluding Residual Activity Returned from Feedwater Turbine). | 287 |
| Hotwell (Excluding Residual Activity Returned from Feedwater Heaters and Gland Seal System) | 18 |
| SJAE First Stage System (5) | .6 |
| SJAE Off-gas System | .4 |
| Gland Seal System (6) | 1.0 |
| F.W. Turbine System (6) | 8.8 |
| Total | 1022.0 |

Notes

(1) 6-Flow machine.

(2) Includes inventory in liquid and steam in reheat tubes and in steam supply line.

(3) Includes total inventory beyond reheater outlet.

(4) Includes total inventory beyound extraction point. Distribution of this will depend on equipment arrangement and sizing.

(5) Includes inventory in steam supply line.

(6) Includes total inventory beyond inlet at steam supply line.

Table 3., Turbine equipment typical total and net
$^{16}N$ inventories (Ci) for a 1200 MWe plant.

| COMPONENT | TOTAL | ABOVE OPERATING FLOOR | |
|---|---|---|---|
| | | GROSS | NET EQUIVALENT |
| Main Steam Lines | 260 | 5 | 1.6 |
| HP Turbine | 6 | 6 | 0.3 |
| HPT to MS/R Piping | 60 | 2 | 1.3 |
| MS/R | 220 | 150 | 21 |
| MS/R to LPT Piping | 17 | 17 | 9 |
| LP Turbines | 10 | 10 | 0.5 |
| FW Heaters & Extraction | 130 | — | — |
| Condenser | 290 | — | — |
| Hotwell | 18 | — | — |
| SJAE & Gland Seal | 2 | — | — |
| FW Turbine | 9 | — | — |
| | 1022 | 190 | 34 |

Table 4. Summary of Shielding Cost Estimates

| Inside panel | MSRH cover | Turbine cover | calculational models | | field measurements | | Industry (5)* | | EPA | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | 100% Occupancy & Capacity | 50% Occupancy 80% Capacity | 100% Occupancy & Capacity | 50% Occupancy 80% Capacity | 1 | 2 | Min. | Max. |
| – | – | – | 33 | 13 | 15 | 6.0 | 'base' | 'base' | 'base' | 'base' |
| 6" | 3" | – | 13 | 5.2 | 5.9 | 2.4 | – | 720 | 96 | 136 |
| 6" | 6" | – | 8.8 | 3.5 | 4.0 | 1.6 | – | 745 | 122 | 169 |
| 1' | 1' | – | 5.9 | 2.4 | 2.7 | 1.0 | 255** | 890 | 205 | 271 |
| 1' | 1' | 6" | 3.0 | 1.2 | 1.4 | 0.5 | | 915 | 258 | 347 |
| – | – | – | 37 | 15 | 17 | 6.7 | 'base' | 'base' | 'base' | 'base' |
| 6" | 6" | – | 16 | 6.4 | 7.3 | 2.9 | – | 745 | 122 | 169 |
| 1' | 1' | – | 8.6 | 3.4 | 3.9 | 1.6 | – | 895 | 205 | 271 |
| 2' | 1' | – | 5.8 | 2.3 | 2.6 | 1.1 | 255 | 990 | 257 | 327 |
| 2'3" | 2' | 6" | 2.8 | 1.1 | 1.3 | 0.5 | – | 1,250 | 348 | 469 |

\* Two estimates were provided, both are shown
\*\* This cost thought to represent an option [inbetween] the final two in this category
Note: First five options for turbine perpendicular to boundary, second five options for turbine parallel to boundary.

BWR TURBINE BUILDING LAYOUT WITH
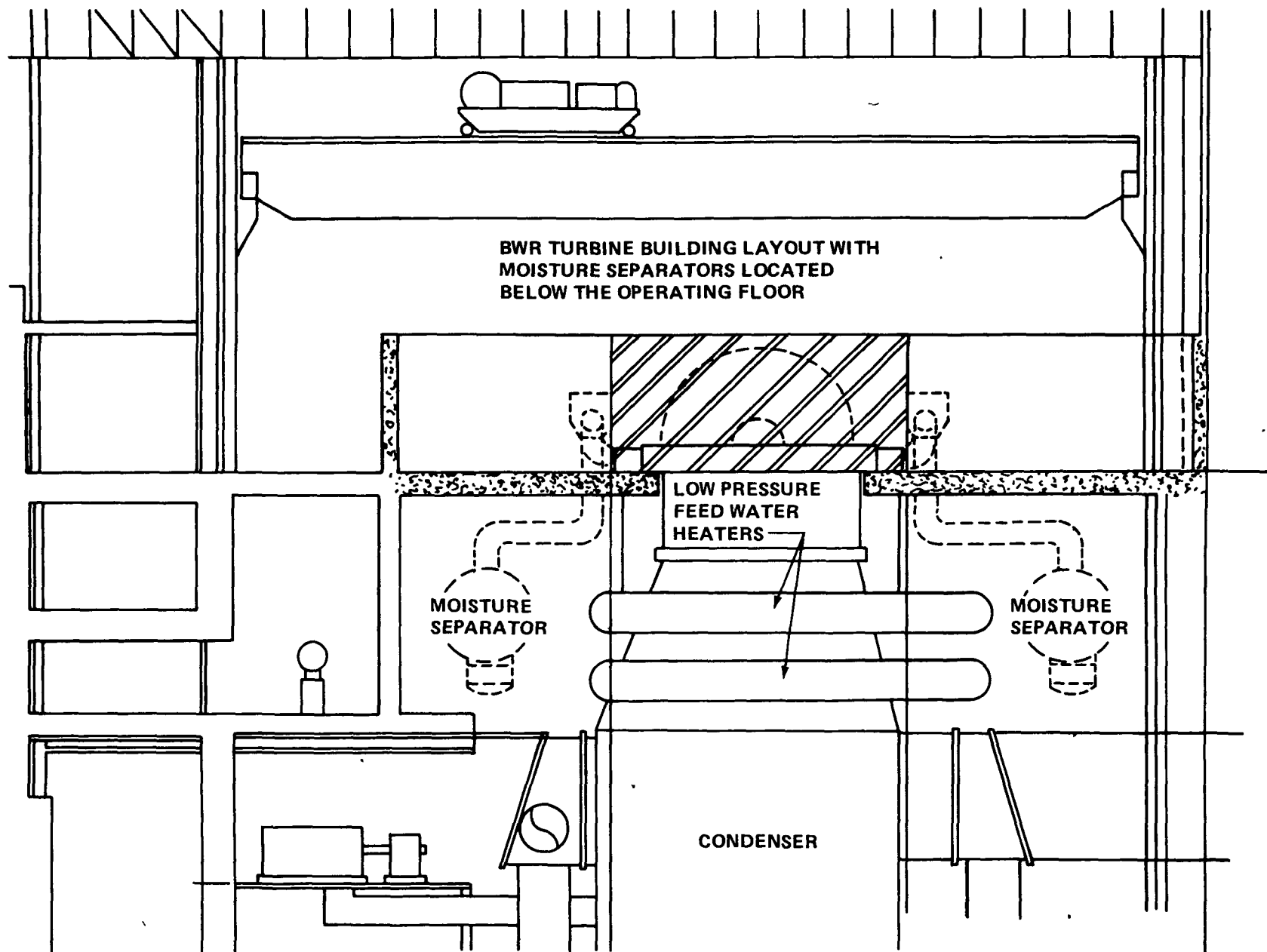MOISTURE SEPARATORS LOCATED
BELOW THE OPERATING FLOOR

LOW PRESSURE
FEED WATER
HEATERS

MOISTURE
SEPARATOR

MOISTURE
SEPARATOR

CONDENSER

FIGURE 1. TYPICAL COMPONENT LAYOUT IN EARLY BWR TURBINE BUILDING DESIGNS.[5]

ROOF SLAB LOCATION WHEN USED

INSIDE PANEL WHEN USED

MOISTURE SEPARATOR

MOISTURE SEPARATOR

26'

26'

19.5'

LOW PRESSURE FEED WATER HEATERS

CONDENSER
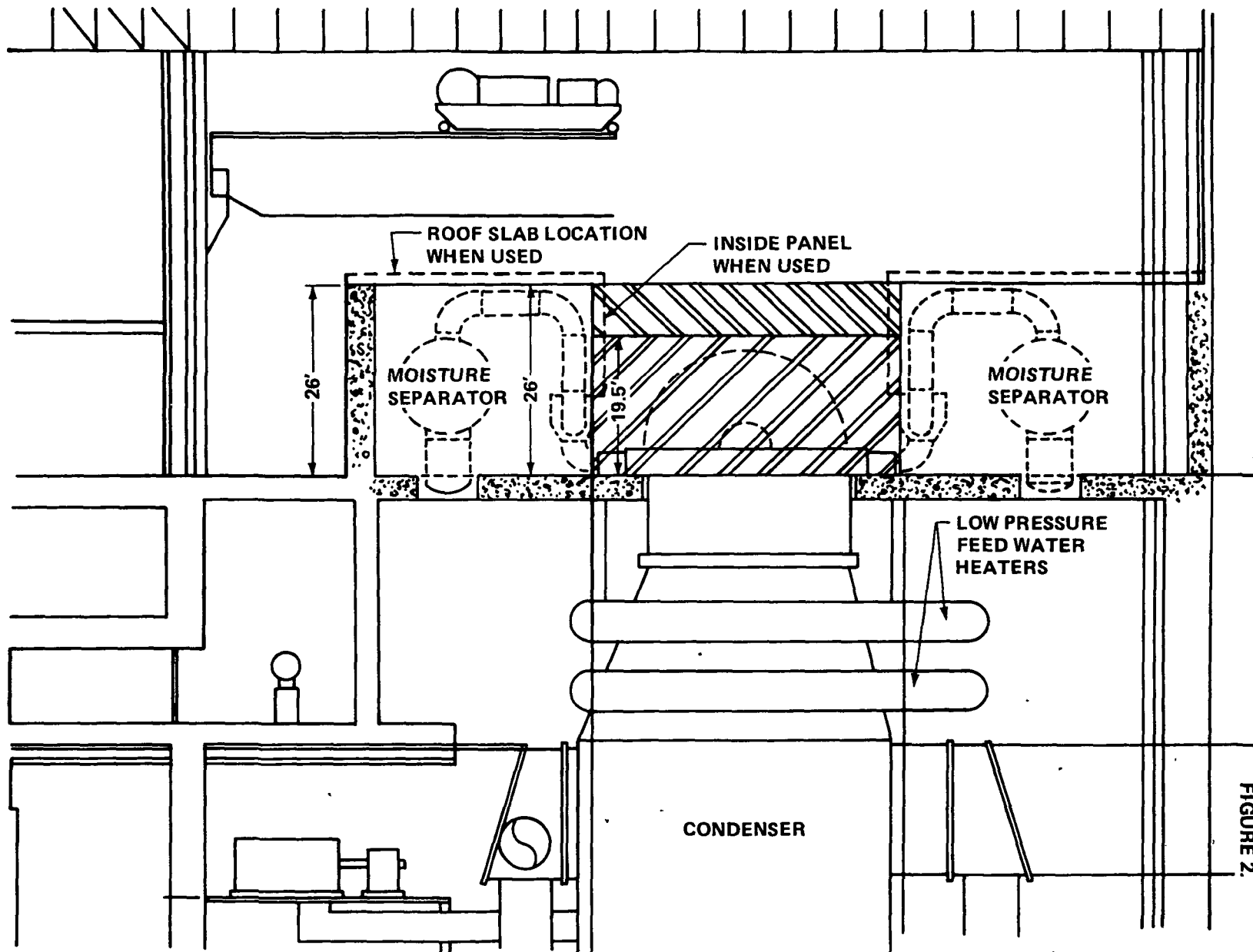
FIGURE 2.

FIGURE 2. TYPICAL COMPONENT LAYOUT IN CURRENT BWR TURBINE BUILDING DESIGNS.[5]

AIR—SCATTERED

BUILDING
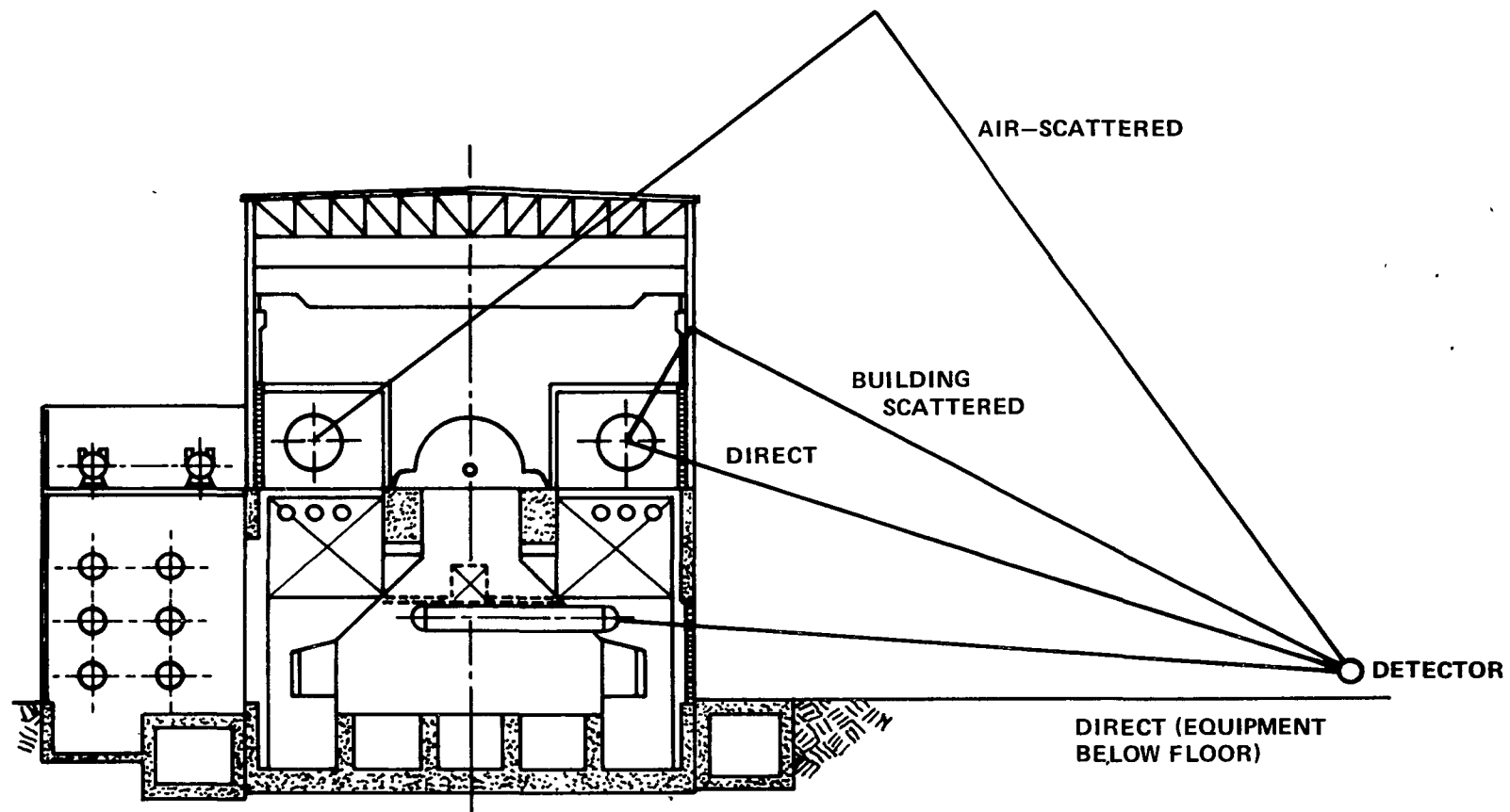SCATTERED

DIRECT

DETECTOR

DIRECT (EQUIPMENT
BELOW FLOOR)

FIGURE 3. CONTRIBUTIONS TO DOSE RATE FROM N—16 IN TURBINE BUILDING COMPONENTS.
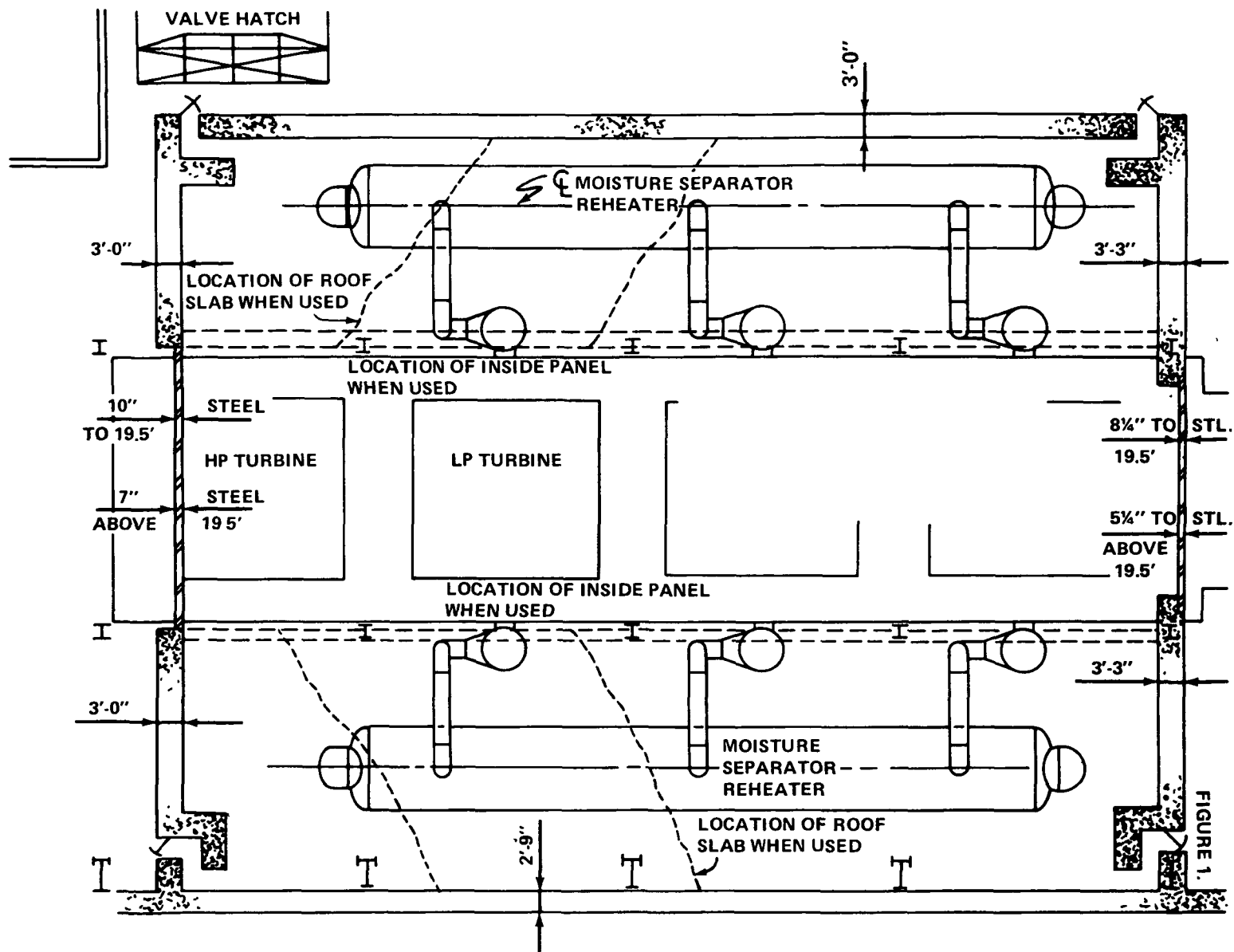
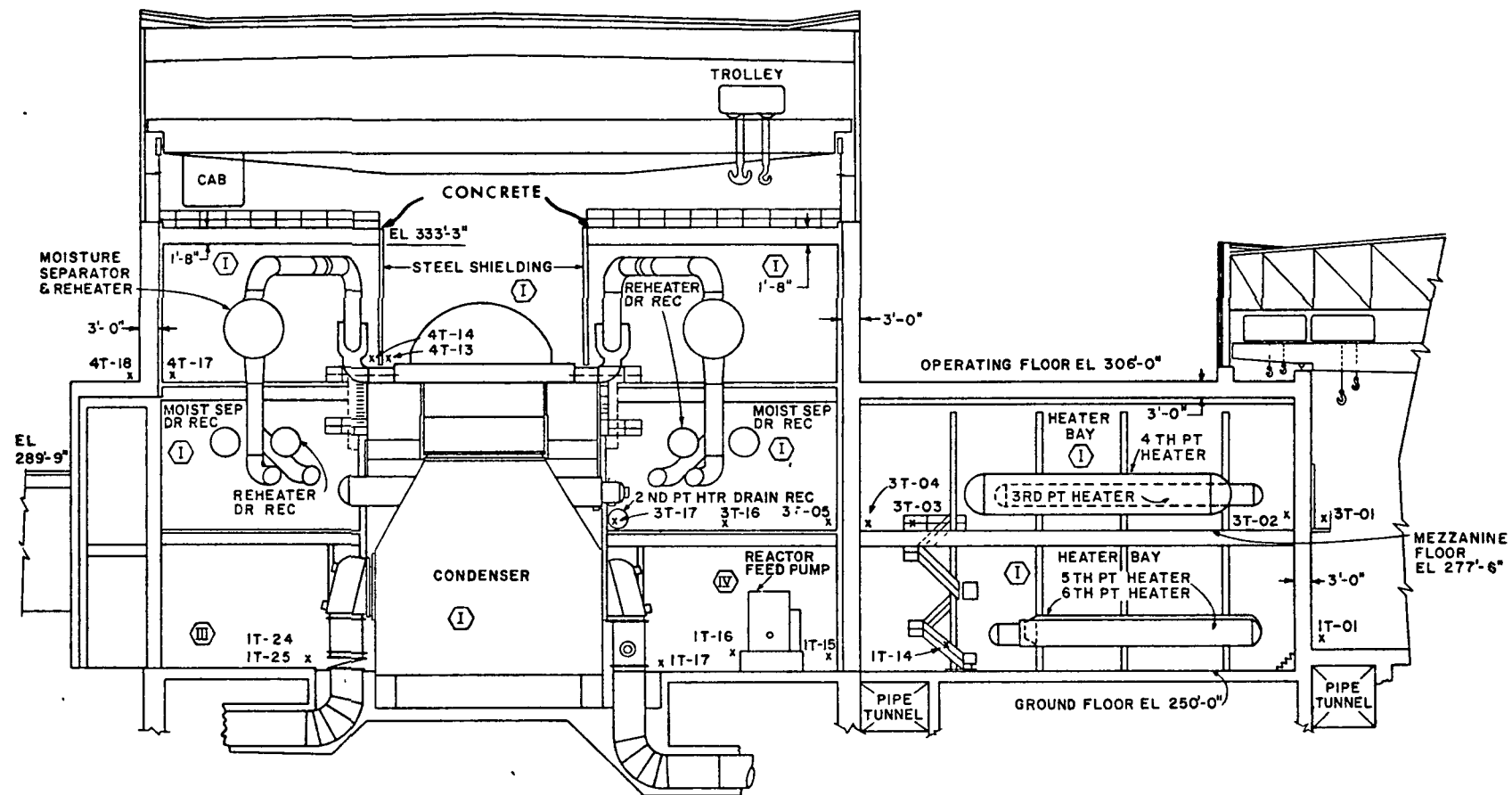FIGURE 4. TOP VIEW OF TURBINE COMPONENT LAYOUT SHOWING TYPICAL "ACCESS" SHIELD DESIGN ALONG WITH VARIOUS SHIELD OPTION.[5]

Figure 5. Transverse sectional view of Nine Mile Point 2 nuclear plant turbine building, showing shielding of moisture separators and turbines.[3]
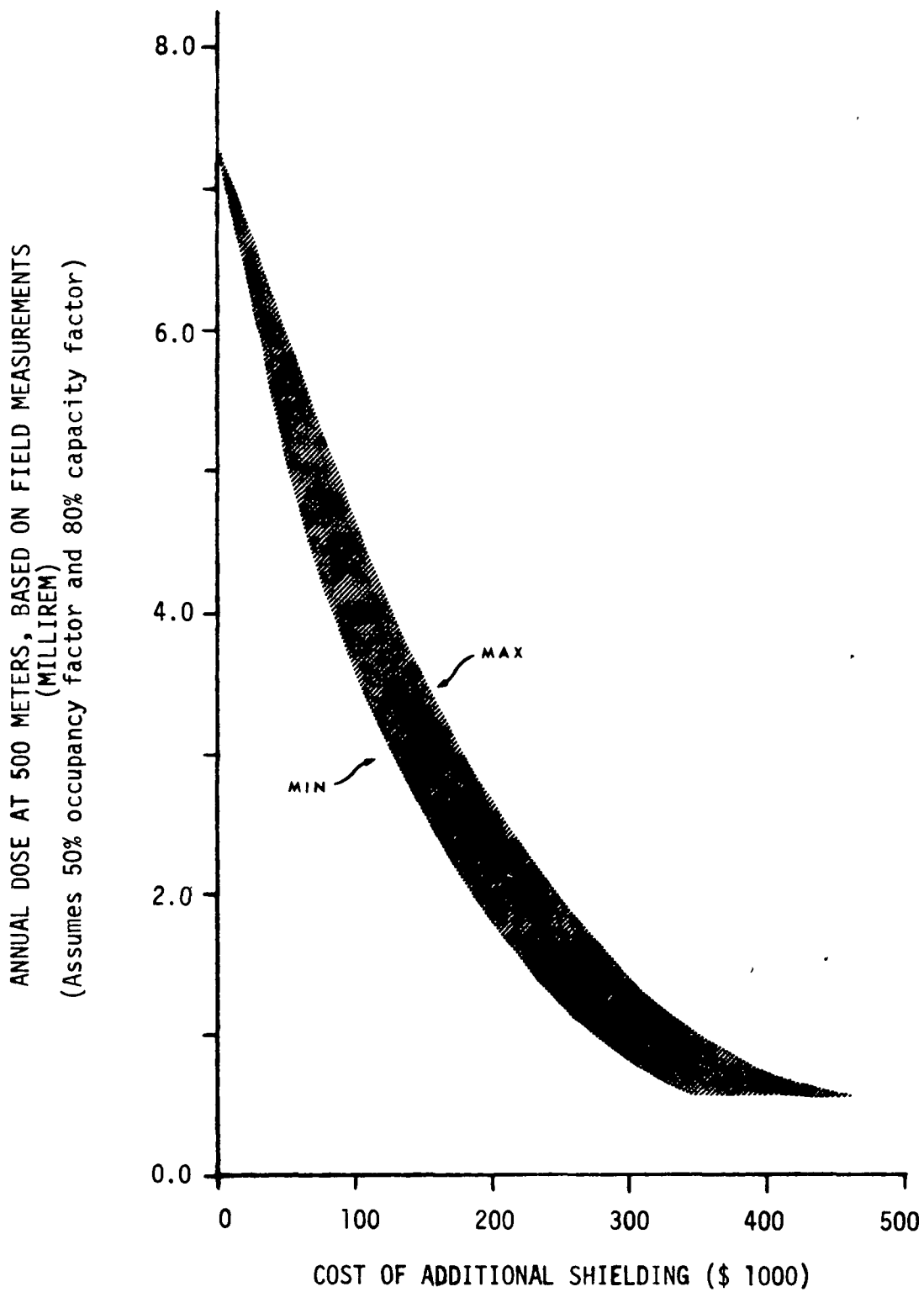
Figure 6. ANNUAL DOSE AT 500 METERS VS. COST OF SHIELDING
(Turbine parallel to boundary)

SUPPLEMENT    E


THE PROPOSED STANDARDS AND THE

NUCLEAR ENERGY CENTER CONCEPT

# EPA PROPOSED STANDARDS FOR THE URANIUM FUEL CYCLE
## AND THE NUCLEAR ENERGY CENTER CONCEPT

## Introduction

The Federal Register notice proposing these environmental radiation
standards for the uranium fuel cycle pointed out that "...in view of the
need to accumulate operating experience for the new large individual
facilities now under construction and the intent of the Agency to review
these standards at reasonable intervals in the future, it is considered
premature and unnecessary to predicate the standards on any siting
configurations (e.g., nuclear energy centers) postulated for the next
decade and beyond. The Agency will consider changes in these standards
based on such considerations when they are needed and justified by
experience..." (1). The proposed standard does not itself specify
standards for any specific siting configuration, nor is any siting
concept excluded from its applicability. EPA's commitment is simply to
reconsider the standard when data is available on which to base an
evaluation of the nuclear energy center (NEC) concept.

A number of commenters on the proposed standards addressed the NEC
concept in somewhat general terms. They expressed two types of concerns.
The first was expressed by one commenter as follows: "...however, the
proposed limits may discourage plans for energy parks for the following
decades. Since the (sic) energy parks may well offer reduced overall
radiation and health effects to the general public (at the expense of

slightly higher individual exposures) along with possible cost savings and safeguards improvements, the long range implications of the standards on the parks should be explictly addressed..." (2). The second concern seen is: "By specifically excluding nuclear parks from the standards, EPA makes utility planning for the design, purchase and construction of future nuclear power plants difficult" (3). None of the commenters provide any quantitative information to support their concerns.

## Background

Only a few studies of nuclear energy centers have been published. One, titled "Assessment of Energy Parks vs. Dispersed Electric Power Generating Facilities," and sponsored by the National Science Foundation (4), did not treat radioactive effluents in enough detail to indicate whether the proposed standards could or could be met. That study referenced "Evaluation of Nuclear Energy Centers" (WASH 1288) on this matter (5).

WASH 1288 provides the most complete treatment of NEC's available to date, and evaluates two real sites in enough detail to draw some conclusions (albeit imprecise) prior to the more detailed studies of the NEC concept now almost completed by NRC. Appendix 1 of WASH 1288 provides a discussion of the Hanford reservation in Richland, Washington as a potential site, which includes an evaluation of potential radioactive effluents. The results indicate that 25 reactors and a reprocessing plant could be sited at Hanford with a radiological impact

which should be significantly less than permitted by the proposed standards (6).

Appendix 2 of WASH 1288 provides a similar treatment of a site at River Bend, Louisiana, and also estimates an impact less than that permitted by the proposed standards (7). It should be noted that WASH 1288 was written in 1973, and the authors were concerned with meeting the then proposed Appendix I. Thus, effluent controls are assumed in the discussions that will achieve calculated doses in accordance with proposed Appendix I.

Appendix 5 of WASH 1288, "Radiological Impact of a Nuclear Center on the Environment" contains a generic treatment of radioactive effluents by Soldat. Based on his evaluation, it appears that the proposed standards for atmospheric releases would be met if prudent site selection is made and reasonable levels of effluent control provided.

One potential problem indicated by Soldat that would require special attention is liquid releases. If radionuclides are released from a large number of reactors into a single body of water, special radwaste or operating procedures may be necessary, such as onsite receiving ponds. This would depend on the specific characteristics of the water body for receiving possibly large quantities of radionuclides (8).

WASH 1288 does not answer all of the concerns expressed by commenters on the proposed standards. Existing analyses are of a scoping nature and do not address the advantages and disadvantages of NEC's versus dispersed siting, nor in any detail the impact of other considerations (thermal and potential accidents, for example), which

would certainly be appropriate to any decision on standards specifically
designed for NEC's.

The "Nuclear Energy Center Site Survey" (NECSS) now underway by the
NRC is expected to provide much of the data and analysis necessary to
make a sound decision on the viability of the NEC concept. A number of
surrogate sites, as well as hypothetical sites, will be analyzed and
various combinations of reactors and fuel cycle facilities will be
considered. It is EPA's understanding that the NRC staff conducting this
survey has considered these proposed standards and the associated DEIS
during its study of potential NEC's. An examination of the preliminary
results of the NRC study does not reveal any significant conflicts
between the proposed standards for the uranium fuel cycle and the
feasibility of the NEC concept. Such a preliminary finding does not, of
course, preclude a later finding, based on a more detailed study, that
some specific provisions may be required in the standards for such sites.

Discussion

The task of completely assessing the potential impact of the
proposed standards on NEC's is beyond the scope of this discussion.
However, some of the unique aspects of NEC's that are involved can be
briefly mentioned.

There are some characteristics of NEC's that will make doses to
members of the public less than might be expected on the basis of
assessments for conventional sites. The exclusion distance or the
distance to the nearest boundary from such a large group of plants can be
expected to be greater than for smaller numbers of facilities on
conventional sites. A distance of one to one and one half miles may be

typical versus the typical one half or less miles for conventional sites.
The sites for NEC's are likely to be quite large (50-75 square miles)
with the plants dispersed over the site in order to minimize effects from
thermal releases to the atmosphere. NEC sites may also be relatively
remote. Economies of scale and shared systems may also make some
effluent control systems available that would not be cost-effective at
conventional sites.

The dose at the site boundary will not be the multiple of the number
of reactors times the dose from the nearest reactor to the site boundary.
Soldat (8) has calculated that the increase in dose over that due to the
nearest facility (or group) would be a factor of from two to five. A
scoping calculation carried out by EPA for thyroid doses arrives at a
factor of three. Of course this would vary depending on actual site
factors and could increase with the addition of other fuel cycle
facilities, such as fuel reprocessing. However, on a large site one
would expect that such other fuel cycle facilities would be placed well
away from the boundary of the large sites required for NEC's and not
contribute a disproportionate part of the total dose.

Before definitive conclusions can be drawn, all pathways will have
to be considered on a consistent basis; the sensitivity of doses to a
variety of site factors will require evaluation; the effect of adding
fuel cycle facilities must be quantified; quantification of the total
population dose reduction and related benefits achieved by such sites in
relation to any increased maximum individual dose will be necessary; and

any benefits that could be achieved through shared effluent control systems will have to be evaluated.

Based on the information now available, the lack of any other quantitative input from any source to the contrary, and the expectation of prudent and sound siting decisions, it appears unlikely that nuclear energy centers would be unable to meet the proposed standards. However, EPA will review the entire spectrum of analyses of expected impacts and benefits that should be provided in part by the NECSS, in part by future more detailed assessments of specific sites, and in part by experience in the immediate future with existing facilities, in order to arrive at a judgment on the appropriateness of these environmental radiation standards for nuclear power to such possible future siting configurations.

# References

1.  Federal Register, 40, May 29, 1975, p. 23424.

2.  Attachment to letter (H. Hollister, ERDA, to R.E. Train, EPA, September 25, 1975) entitled "Staff comments on proposed EPA regulation (40 CFR Part 190) 'Environmental Radiation Protection Standards for Nuclear Power Operations' and accompanying draft environmental impact statement," p. 6.

3.  Letter, W.D. Crawford, Edison Electric Institute, to Director, Criteria and Standards Division, EPA, July 24, 1975.

4.  Assessment of Energy Parks vs. Dispersal Electric Power Generating Facilities, May 30, 1975, National Science Foundation, NSF 75-500.

5.  Evaluation of Nuclear Energy Centers, January 1974, U.S. Atomic Energy Commission, WASH 1288.

6.  Ibid., Appendix 1, p. 7.24.

7.  Ibid., Appendix 2, p. 7.67 et. seq.

8.  Ibid., Appendix 5, p. 13.

SUPPLEMENT    F


CONTROL OF KRYPTON AND IODINE DISCHARGES

FROM NUCLEAR FUEL REPROCESSING FACILITIES

# CONTROL OF KRYPTON AND IODINE DISCHARGES
# FROM NUCLEAR FUEL REPROCESSING FACILITIES

## I. Introduction

Radioactive krypton-85 and iodine-129 discharges from reprocessing

facilities have chemical and physical properties which make their

collection and retention technically difficult. Krypton is a chemically

inert gas, and iodine is volatile at normal temperatures and pressures.

It has been the practice to discharge to the atmosphere all of the

krypton-85 present in spent reactor fuel. Iodine-129 in spent fuel has

been recognized as a potentially significant environmental contaminant,

and efforts have been made in the past to control the discharge of this

species of radioactive iodine. These efforts were only partially

successful, however, and it has become increasingly apparent that

improved control of long-lived radio-iodine discharges from fuel

reprocessing facilities is necessary (1,2). Current estimates of the

costs and control efficiencies of a variety of improved control systems

for iodine-129 and the most important options for control of krypton-85

are reviewed below. The benefits to be gained by reducing the

environmental dose commitments associated with releases of these

materials through installation of such systems are then set forth.

Finally, the level of cost-effectiveness of each of the control options

is determined.

## II. Source Terms for Iodine and Krypton

The quantities of iodine-129, iodine-131, and krypton-85 present in spent fuel have been previously reported, based on calculations using the computer code ORIGEN (3). These values, expressed in curies per metric ton of heavy metal in the fuel, are:

Kr-85:     10,500 Ci/MTHM

I-129:      0.4 Ci/MTHM

I-131:      0.9 Ci/MTHM

for the following fuel parameters, used in this report:

Burnup = 33,000 MWd/MTHM

Average Specific Power = 30 MW/MTHM

Cooling Time = 160 days.

It is assumed that a light-water-cooled power reactor operates at 33% thermal efficiency, producing approximately 33 MTHM of spent fuel with this burnup for each gigawatt-year of electric power[GW(e)-yr], and that a typical fuel reprocessing plant has a throughput capacity of 1500 MTHM per year. Such a plant would be capable of processing the spent fuel from about 45 such reactors each year.

If no iodine or krypton control systems were installed at a 1500 MT plant, the number of curies discharged annually would be:

Kr-85:     16,000,000 Ci

I-129:          60 Ci

I-131:       1,400 Ci

It is assumed that these contaminants are discharged to the atmosphere, rather than into liquid pathways, since currently projected plants use

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complete recycle of process liquids and thus no liquid discharges are planned.

Although the source term for I-131 could theoretically approach 1400 Ci per year, it is highly unlikely that such quantities will be available for discharge in actual operations because of its relatively short half-life (8.08 days). Even if all spent fuel was process at 160 days cooling time, any delay of iodine-131 in the various inplant processes or off-gas streams would permit additional decay and reduce the quantity available for discharge. Other factors that would reduce the quantity of iodine-131 available for discharge include: a) the existing large backlog of spent fuel, which indicates there is no need, at least in the foreseeable future, to process fuel that has been cooled for only 160 days, b) cooling requirements for spent fuel shipping casks may be such that the fuel cannot be loaded for shipping from the reactor to the reprocessor until it has cooled for periods greater than 160 days, and c) for those reprocessing plants using in-line solidification of high level waste, cooling periods in the range of a few years may be required to permit sufficient decay of radioactive ruthenium. Thus, it is considered highly unlikely that the I-131 source term at fuel reprocessing will approach the theoretical maximum value.

III. Control Technologies for Krypton at Reprocessing Plants

Since krypton is a chemically inert noble gas, it follows the process off-gas stream in the fuel reprocessing plant and will be discharged to the atmosphere unless specially designed air-cleaning systems are used to capture it. Standard air-cleaning systems based on

chemical processes are ineffective in collecting noble gases. Most of

the krypton produced by the fission process in the reactor is released to

the off-gas stream during dissolution of the spent fuel (4,5). A small

fraction is also released during the shearing operation, but this

fraction is also routed to the main off-gas stream. Thus, all of the

krypton-85 present in the spent fuel is collected in one stream, along

with other contaminants, such as oxides of nitrogen, hydrocarbons, and

other radioactive materials.

Two basic systems are in advanced stages of development for the

control of krypton-85; the the cryogenic distillation system and the

selective absorption system. These are discussed in turn, briefly,

below:

### 1. Cryogenic Distillation

This process is widely used in industry, where it is better

known as the "liquid air" process and is used to condense and separate

the various gaseous components of air. Heat is removed from air in the

gaseous form in a closed system until the boiling points of the various

gaseous components are reached. As the boiling point of each component

is reached, it liquifies and can be separated from the remaining gaseous

components having lower boiling points. Since krypton has a boiling

point of minus 224 °F and the two major constituents of air, nitrogen and

oxygen, have boiling points of minus 322 °F and minus 297 °F,

respectively, liquifaction and separation of the krypton poses no serious

technical problem. Several descriptions of applications of such systems

to nuclear power plants are available (6-13).

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The most serious potential difficulty associated with cryogenic

systems is the possibility of explosions due to a buildup of hydrogen,

acetylene, hydrocarbons, and oxygen (or ozone) in the system (10). This

can be avoided by chemically removing all oxygen before the gas stream is

introduced into the cryogenic apparatus (6). Thus, in order to use this

process, two additional systems are required: a) a catalytic converter

system to convert oxygen to water and carbon dioxide, followed by, b) a

system for removal of these products. While this entire system has not

yet been reduced to commercial practice through demonstration in an

operating fuel reprocessing plant, on the basis of existing laboratory

and pilot plant experience it appears feasible for such use and is

expected to be available by 1983 (10). In addition to determining that

the explosion potential of the cryogenic systems is effectively removed

by precleaning the gas stream following use of a catalytic converter, a

full assessment of the remote operation and maintenance capabilities of

this system must be completed in the interim.

The cryogenic system itself is expected to exhibit a

decontamination factor (DF) of at least 1000 (6). However, the overall

efficiency for removal of krypton from the plant is expected to be

somewhat lower because of potential leakage through the system during

startup and shutdown operations, maintenance, etc. Therefore, an

effective plant DF of between 10 and 100 has been projected for routine

operation of such a system (14).

## 2. Selective Absorption

This process was developed at the Oak Ridge National Laboratory (ORNL), specifically for the control of krypton-85 at fuel reprocessing plants (15,16). The process is based on preferential dissolution of noble gases in a fluorocarbon sorbent, such as the refrigerant freon-12. The off-gas stream is passed through the sorbent in an absorber column at a relatively low temperature and high pressure. Essentially all of the krypton and xenon present are dissolved in the sorbent, along with other components of the gas stream. The other components are then removed in a fractionating desorption system and, essentially free of krypton and xenon, recycled to the off-gas stream. The sorbent is then transferred to a stripper system where a product gas concentrated in krypton and xenon is evolved and collected. The pure sorbent is then regenerated and returned to the absorber column.

The selective absorption process has exhibited a decontamination factor greater than 1000 in tests with nitrogen oxides and carbon dioxide (10). However, further investigations are expected to be accomplished to define the relevant auxiliary systems required for successful application of the selective absorption process. An effective DF of between 10 and 100 has been conservatively designated for this process. The selective absorption system is free from explosion and fire hazards, and can be operated routinely for sustained periods with low maintenance requirements. This system has also not been demonstrated at an operating commercial reprocessing plant. However, it has been offered commercially for use on the gaseous effluents from nuclear power

reactors (17). A recent review concluded that additional testing is required for this system using off-gas containing significant amounts of contaminants and estimated that the process could be reduced to practice by 1983 (10).

Estimated costs for installation and operation of these two systems are listed in Table 1. Cryogenic distillation system costs are based on industry estimates of equipment costs (18,19) corrected to installed costs. Estimated costs for the selective absorption system are based on ORNL estimates of equipment costs (10).

In order to satisfy the proposed standards, storage for 40-70 years would be required, depending upon the degree of initial decontamination achieved, in order to insure adequate decay. The management of krypton-85 following its collection has been addressed by Foster and Pence (20) and appears to present no serious problems. They reviewed the advantages and disadvantages of long-term storage of krypton-85 in high pressure steel cylinders and concluded that this appears to be a practical method for fission-product noble gas storage. Final storage of krypton-85 could take place either at the fuel reprocessing facility or at a properly designed central waste repository.

IV. Control Technologies for Iodine at Reprocessing Plants

The control of iodine at reprocessing plants is a significant technical challenge (7). During the last few years a number of promising systems for control of iodine in gaseous waste streams have been investigated and most are now in various stages of final demonstration for commercial use. The principal remaining problem, as pointed out in

the previous EPA report concerning fuel reprocessing (1), is that, until recently, inadequate attention has been given to the control of iodine in low-level liquid waste streams. Any iodine present in these liquid streams, whether from off-gas scrubber solutions or from other sources, can potentially be discharged to the environment because of its high volatility. Evaporative processes are used to reduce the volume of these low-level liquid wastes and to provide for discharge of tritium to the atmosphere. Such processes will, of course, also drive off any iodine present for subsequent discharge to the atmosphere, and systems developed for removal of iodine from gaseous streams are not, in general, applicable to evaporator discharges because of their high water content.

A simplified schematic of waste streams appropriate to the discussion of iodine control systems for current designs of reprocessing plants is shown in Figure 1. Most of the iodine present in spent fuel is released to the off-gas system during the fuel dissolution and initial processing steps. The fraction released to the off-gas has been estimated at no less than 90% (21). The balance is collected in liquid waste streams. The off-gas system for a specific plant will not necessarily be designed just as shown in the schematic, since the detailed design can vary due to the order in which contaminants are removed. For example, it may be advantageous to remove the oxides of nitrogen from the dissolver off-gas stream before dilution by process off-gas inputs.

Table 2 summarizes iodine control system capabilities and costs. The iodine control system DF's assumed are, for the most part, those used

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in a recent study of effluent controls for fuel reprocessing by
ORNL (10). It should be noted that there are differences between the
estimates of systems performance in the ORNL report and those presented
in testimony at a recent licensing hearing for the Barnwell fuel
reprocessing facility (21). In general, the ORNL analysis predicts
higher DF's for off-gas systems. For example, the DF's shown on Figure 1
yield an effective overall DF of about 100 for I-129 and about 500 for
I-131. Those presented in the Barnwell hearing (21), in contrast, are
approximately 20 for I-129 and about 40 for I-131 for the plant overall.
That testimony assumes that the mercuric nitrate scrubber bottoms are
discharged into the low-level liquid stream. The management of these
scrubber bottoms is the major source of the difference between these
estimates. It will be necessary to retain the bulk of iodine in the
scrubber bottoms in order to achieve effective control of iodine.

The difference in control efficiencies for I-129 and I-131 shown in
Table 2 for Ag-Z and macroreticular resins are due primarily to the
differences in half-lives of these radionuclides, as discussed in detail
by Davis (22). This difference is to be expected in any system which
relies upon delay as part or all of its operating principal. Thus, it is
essential to both isolate and contain long-lived radionuclides to insure
that they will not eventually re-enter a discharge stream.

The chemical form or species is an important characteristic of the
iodine when considering cleaning efficiencies, environmental transport,
and iodine dosimetry. In general, it is believed that iodine evolved
during the dissolution process will be in the elemental form (23).

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However, any iodine discharged to the off-gas system during or following the separation processes is considered likely to have a large organic component (24). The relative fractions of iodine evolved from the dissolution process step and from the various subsequent separation processes is not known, nor is the organic component of either fraction (21). Estimates of these fractions vary widely (21,25) and these differences will probably not be resolved until studies are conducted during actual operations of a large facility (25). For the purposes of this analysis it is assumed that 90% of iodine is discharged to the off-gas system, with the balance going to liquid waste streams (21). The fraction of the iodine discharged to the atmosphere following all control systems is assumed to be about 50% organic and 50% elemental. Factors contributing to an expectation of a significant organic component of the final discharges are: a) iodine from the low-level liquid pathway has passed through organic processing steps and thus can be expected to have a significant organic component, b) iodine in the off-gas stream is expected to contain a significant organic contribution from separation processes, and c) most iodine cleaning systems are more efficient in removing elemental than organic iodine, and thus selectively allow passage of organic iodides.

A brief description of each of the iodine control systems considered follows:

1. Caustic Scrubbers

Caustic scrubbers are widely used in the chemical industry to remove contaminants from off-gas streams (26). They have been used in

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the nuclear industry to control both ruthenium and iodine (27). Tests have indicated that DF's of 100 and greater for elemental iodine are attained (27), but DF's are less for organic iodine species. The fraction of organic iodine in the primary off-gas stream is not known, but is predicted to be low (21). It has been assumed that the organic fraction is less than 10% and that caustic scrubbers will, therefore, operate routinely with a removal efficiency of no less than 90%. Cost estimates for a caustic scrubber are abstracted from the ORNL work (10).

2. Mercuric Nitrate Scrubbers

Mercuric nitrate-nitric acid scrubbers have been used at the AEC (now ERDA) reprocessing facilities at Idaho Falls, as well as at a commercial facility (Nuclear Fuel Services) to control the discharge of iodine. While this type of scrubber removes both elemental iodine and organic iodides, tests have indicated that it is also more efficient in removing iodine in the elemental form (28). Based on the predicted relative fractions of organic iodides present (21), it is assumed to remove about 90% of all iodine from the off-gas stream (28,29). Costs for mercuric nitrate scrubbers are expected to be similar to those for caustic scrubbers (1).

3. Silver Zeolite Adsorbers

Silver zeolite adsorbers have not been used to treat reprocessing plant off-gas, but are scheduled to be installed in future plants. Most of the development work for this system was conducted at Idaho National Engineering Laboratories Falls (30). Silver nitrate is impregnated into an alumina-silica matrix and the resulting material is

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arranged in a relatively deep bed, since a longer residence time of the iodine in the adsorber appears to enhance its efficiency. High removal efficiencies have been observed for all chemical species of iodine using this process (30). Although considerably higher values are reported for small-scale systems, ORNL assigned a DF of 10 for I-129 and a DF of 100 for I-131 for a silver zeolite adsorber, pending the development of additional data for plant-scale usage (31,32), and these conservative values have been assumed here. The costs, which are abstracted from previous EPA work (1), are subject to some uncertainty related to the loading rate of the system and thus the quantities of silver required.

4. Macroreticular Resins

Adsorption of iodine from both neutral and slightly acidic solutions on macroreticular resins has been shown to be about 99% efficient in laboratory studies (33). However, performance of this system has not been demonstrated in commercial-scale practice and, until proven under operating conditions, a conservative DF of 10 for I-129 and a DF of 100 for I-131 are assigned. Costs for this system are estimated to be small (10).

5. Suppression in Evaporator by Mercuric Nitrate

Mercuric nitrate, when added to liquid evaporators, will suppress the evolution of iodine into the overheads. The Barnwell Facility includes provision (34) for this method of iodine emissions control from liquid waste streams. Yarbro has estimated a DF of 2 to 10 across the final vaporizer for this addition (21). A conservative value

of 2 is assumed for this analysis. Costs are estimated to be similar to those for a macroreticular resin system.

6. Advanced Systems

Figure 2 displays a simplified schematic of an advanced iodine control system. The basic principle of this system is to force essentially all of the iodine into the off-gas system so as to avoid the difficulty of removing iodine from liquid streams, and then to use highly efficient systems to remove and retain iodine from the off-gas. In the schematic this objective is achieved by using an iodine evolution process at the dissolver to drive the iodine into the off-gas, and the Iodox system to efficiently remove the iodine from the off-gas. The voloxidation step is primarily used for tritium control. However, a significant fraction of both the iodine and krypton present in the spent fuel will also be driven off by this process. After tritium has been removed from the voloxidation off-gas, this stream is routed to the dissolver off-gas stream for subsequent krypton and iodine removal.

The Iodox process itself effectively scrubs both elemental and organic iodine from off-gas streams with concentrated ($\sim$20M) nitric acid (23,35). Laboratory-scale studies have indicated that DF's in excess of 10,000 for methyl iodine have been obtained in multi-staged bubble-cap columns (24). The efficiency with which iodine is scrubbed from off-gas streams with nitric acid is dependent on the oxidizing power of the concentrated nitric acid, which converts the volatile iodine species to the nonvolatile $HI_3O_8$ form. The cost estimates in Table 2 are abstracted from the ORNL work (10); there is no provision made at this

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time for the additional cost of a fractionation/system to permit recovery of the acid at low concentrations for recycle to the dissolver and Iodox systems.

The Voloxidation process effectively removes such volatile fission products as iodine and krypton from sheared fuel, by heating the fuel to about 550 °C in air or oxygen to release these fission products by thermal evolution or by oxidation (36,37). The process equipment would consist of: a) a rotary kiln to oxidize the fuel, b) a recombiner to form tritiated water, and c) a drier to collect the water and separate it from iodine and krypton which then flow to the Iodox equipment (36,37). Laboratory-scale tests with highly-irradiated sheared fuel show that up to 75% of the iodine and 45% of the krypton are volatilized. The costs shown are based on the ORNL work (10).

ORNL is currently conducting development work on these advanced systems. Cost estimates and projected DF's are abstracted from their recent summary (10). ORNL has projected that these systems will be demonstrated and available for installation in new reprocessing plants by about 1983.

V.  Cost Evaluations

Estimated capital costs and annual operating costs for the various krypton and iodine control systems described are listed in Tables 1 and 2. For those systems for which only equipment cost estimates were available, a factor of 1.49 times the equipment cost was applied to estimate total capital cost. This factor includes engineering,

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construction, installation, quality assurance, miscellanous, contingency
and some interest costs (10).

The total annual cost listed in Tables 1 and 2 is the sum of the
annual operating costs and the annualized cost of capital. An annual
fixed charge rate of 18% was used to calculate annual fixed charges.
This rate is based on the following assumptions:

| | | |
|---|---|---|
| Plant (equipment) lifetime | 20 years | |
| Capital investiment in bonds | 30% | |
| Capital investment in equity | 70% | |
| Interest rate on bonds | 5% | |
| Rate of return on equity (after taxes) | 16% | |
| Local property tax rate | 3.2% | |
| Annual cost of replacements | 0.35% · | |
| Annual property insurance rate | 0.25% | |

The annual fixed charge rate was calculated as:

| | | |
|---|---|---|
| Return on Investment | = | 12.7% |
| Sinking Fund Factor at 10% | = | 1.75% |
| Miscellaneous | = | 3.8% |
| Annual Fixed Charge Rate | = | 18.25% |

This value is lower than that calculated by ORNL (10), which was based on
a series of earlier cost evaluations (38,39). A review of these
evaluations indicates that economic conditions have changed sufficiently
to warrant the use of the revised rate caOculated above (41). In
particular, the debt-equity ratio has significantly increased during the
past decade and equity returns have decreased proportionally thus

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producing lower annual fixed charge rates, although interest rates on

bonds may be higher than the rate used above. Further, the investment

tax credit for pollution control equipment allows for increased benefits

to the reprocessing  industry which installs the various krypton and

iodine control systems described in Tables 1 and 2 (41). Other tax

advantages that industry receives by investing in such systems are fast

tax writeoffs for depreciation (42), all incurred state and local taxes

(42), some expense-oriented outlays such as insurance (44), and favorable

treatment for adjustments to the capital basis of equipment (45). These

tax considerations, recent trends toward low rates of return on equity

(considerably less than 16%) for those industries that have nuclear

reprocessing interests, and their higher leverage investment status

(higher debt-equity ratios) would also tend to reduce the annual fixed

charged rate below the 18% rate calculated above, which ignores these

additional factors. Finally, a discount rate of 10% was used for both

sinking fund and present worth calculations.

VI.  Doses and Potential Health Impact Attributable to Krypton and Iodine
     Discharges from Fuel Reprocessing

Cumulative environmental dose commitements to the whole body, lungs,

and the gonads, as well as estimated potential health effects

attributable to release of krypton-85 from a model 1500 MTHM/yr plant are

given in Table 3 for a variety of levels of control efficiency. Plant

startup in 1983 and a useful lifetime of control equipment of 20 years is

assumed. A simple model for krypton transport which assumes immediate

and uniform dispersion into the world's atmosphere was used to estimate

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worldwide doses. Total doses calculated using this simple model agree with results from a more detailed multicompartment treatment described by Machta, Ferber, and Hefter (46,47) within a few percent, although the two models do differ regarding the regional distribution of doses delivered immediately following release. Other parameters, such as population growth and distribution, dosimetry, and dose-effect relationships, were handled as described in the previous analysis (1).

Partial cumulative environmental dose commitments to the thyroid and estimated potential health effects attributable to discharges of iodine-129 from a model 1500 MTHM/yr plant were calculated using the specific activity method (1), and are presented in Table 4. These values represent a partial assessment of the total potential dose and health impact of iodine-129 in that the period of assessment following release of this extremely long-lived material (17 million years half-life) is limited to 100 years. Dose commitments were cumulated for releases over an assumed control equipment lifetime of 20 years commencing in 1983. These partial cumulative environmental dose commitments and their associated health impacts are shown for representative values of overall plant decontamination factors obtainable using the control methods described above. The dose-effect assumptions used were derived from more recent values (47,49) than those used in the original analysis (1): a population age weighted value of 60 thyroid cancers per million rems to thyroid was used.

Health effects may also result from exposure of local populations immediately following release of both iodine-131 and iodine-129, in

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addition to the long-term effects described above. Using methods described previously (1) and short term pathway parameters noted below, it is estimated that uncontrolled release of 1400 Ci/yr of I-131 could result in 35 health effects and the release of 60 Ci/yr of iodine-129 could result in 15 health effects over a 20-year period of plant operation commencing in 1983. These values should be added to those listed in Table 4 to obtain a complete estimate of potential health effects attributable to uncontrolled release of radioactive iodines for the first 100 years following release.

In addition to the population doses and impacts calculated above, maximum potential thyroid doses to individuals may also be significant. Tables 5 and 6 list calculated maximum individual thyroid doses from iodine-129 and iodine-131 discharges for a variety of age groups and release fractions. The values for iodine-131 were calculated using dose conversion factors previously described (48). Dose conversion factors for iodine-129 were based upon those used for iodine-131, corrected for differences in pathway and dosimetry dependent upon half-life and effective energy of decay products (1). It is assumed that 50% of the iodine released is in elemental form and 50% is in organic form, and that $X/Q$ is equal to $5 \times 10^{-8}$ sec/m$^3$. Although specific sites could vary significantly from this assumption, it is expected that site selection criteria for fuel reprocessing facilities will reflect particular attention to minimization of the possibility of dose to the thyroid of nearby individuals.

# VII. Cost-effectiveness Considerations

Table 7 displays the estimated cost-effectiveness of risk reduction of the various options considered for both krypton and iodine control. The cost-effectiveness of both options for krypton control is high, compared to that for typical control systems currently in use in the nuclear power industry, and satisfies the criteria used in judging the reasonableness of the proposed standards (50).

Analysis of the options available for control of iodine is complicated by a) the multitude of alternatives available, and b) the variability of the current stage of development of the different processes. It is clear that iodine evolution and the iodox cleanup process represent the most effective improvements over the basic cleanup of gas streams by scrubbers (with or without backup by Ag-Z) and the cleanup of liquid waste streams by macroreticular resins characteristic of current design practice. Unfortunately, reduction to commercial practice of these systems has not been projected to be completed until 1983. Thus, for the few facilities projected to go into operation prior to that date, utilization of less efficient (and, in the case of Ag-Z, less cost-effective) systems will be necessary. However, with the exception of some secondary systems for liquid cleanup ($HgNO_3$ suppression and, in the case of iodine evolution, macroreticular resin), all of the options display acceptably high levels of cost-effectiveness. It should also be noted that although a second scrubber has apparently greater cost-effectiveness than does Ag-Z, use of the latter system may be preferable due to its anticipated higher level of performance for removal of organic iodines.

Although Table 7 does not display overall plant decontamination factors, it can be seen from Tables 2, 5, and 6 that conformance with the proposed thyroid dose limit of 75 mrem/yr can be readily achieved through use of a variety of combinations of systems exhibiting DF's of 100 or more. However, conformance with the proposed limit of 5 mCi/GW(e)-yr or 1.4 kg/yr for iodine-129 (0.225 Ci/yr from a 1500 MTHM facility) by 1983 will require a plant DF of no less than 300. This would be readily achieved by utilization of iodine evolution followed by the iodox process. Successful achievement of this level of cleanup without use of the iodox process will depend to some extent upon future operating experience with less sophisticated systems. Present estimates of their performance are quite conservative because of a paucity of operating experience, especially regarding their performance with iodine-129. However, it is anticipated and highly probable that DF's greater than 300 for iodine-129 could be achieved by 1983 using appropriate combinations of scrubbers and Ag-Z, since a variety of options are available for improving, if necessary, the conservative levels of performance currently projected. These include a) tandem operation of systems, b) additives, such as thiosulfate to caustic scrubbers, to improve their efficiency (51) c) use of iodine evolution to reduce the fraction of iodine in the liquid waste stream and increase the efficiency of scrubbers by reducing the organic content of the gas streams, and d) demonstration of more efficient cleanup of liquid streams than currently assumed.

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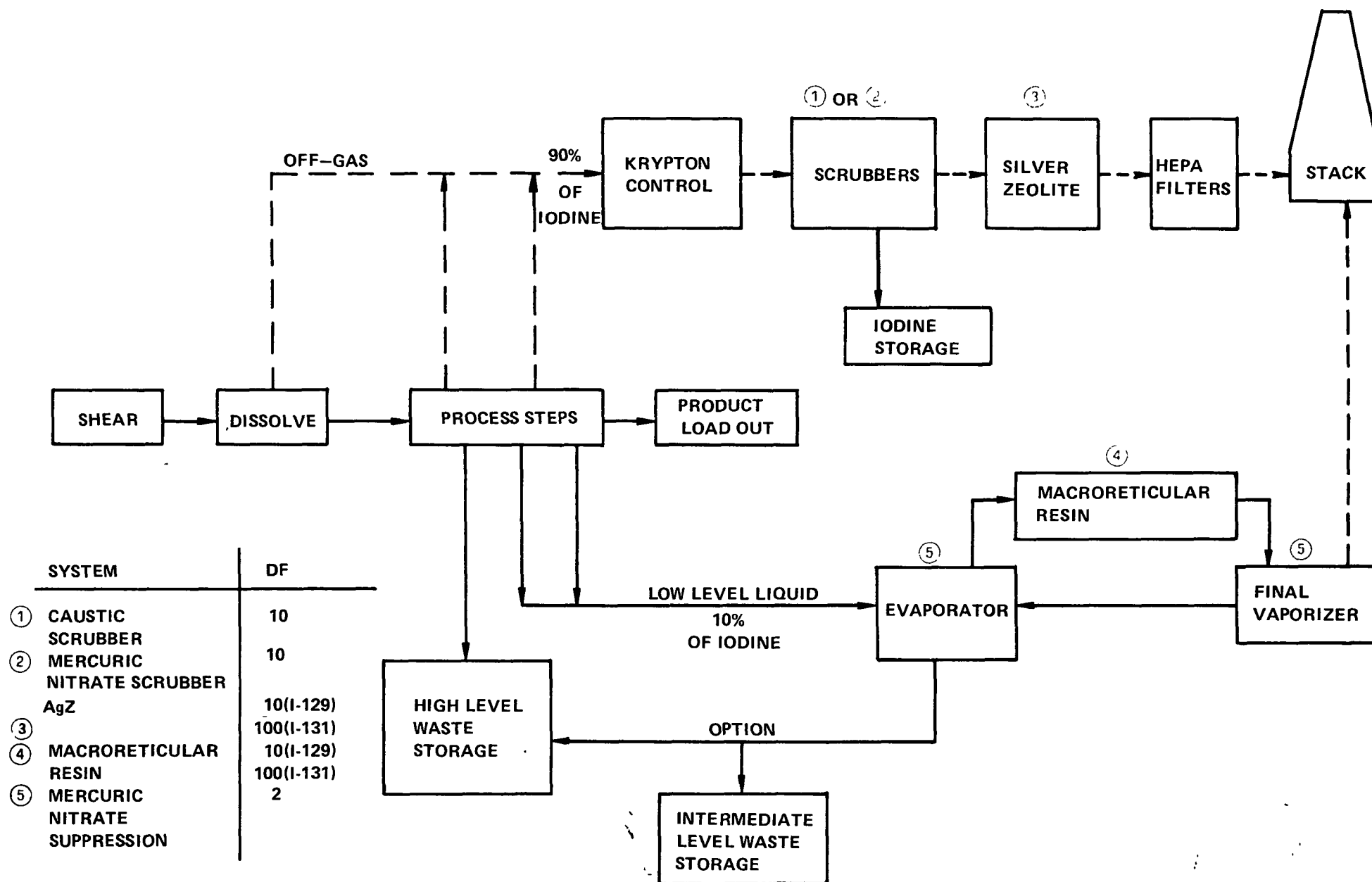
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FIGURE 1. SIMPLIFIED SCHEMATIC OF CURRENT IODINE CONTROL SYSTEMS AT REPROCESSING PLANTS
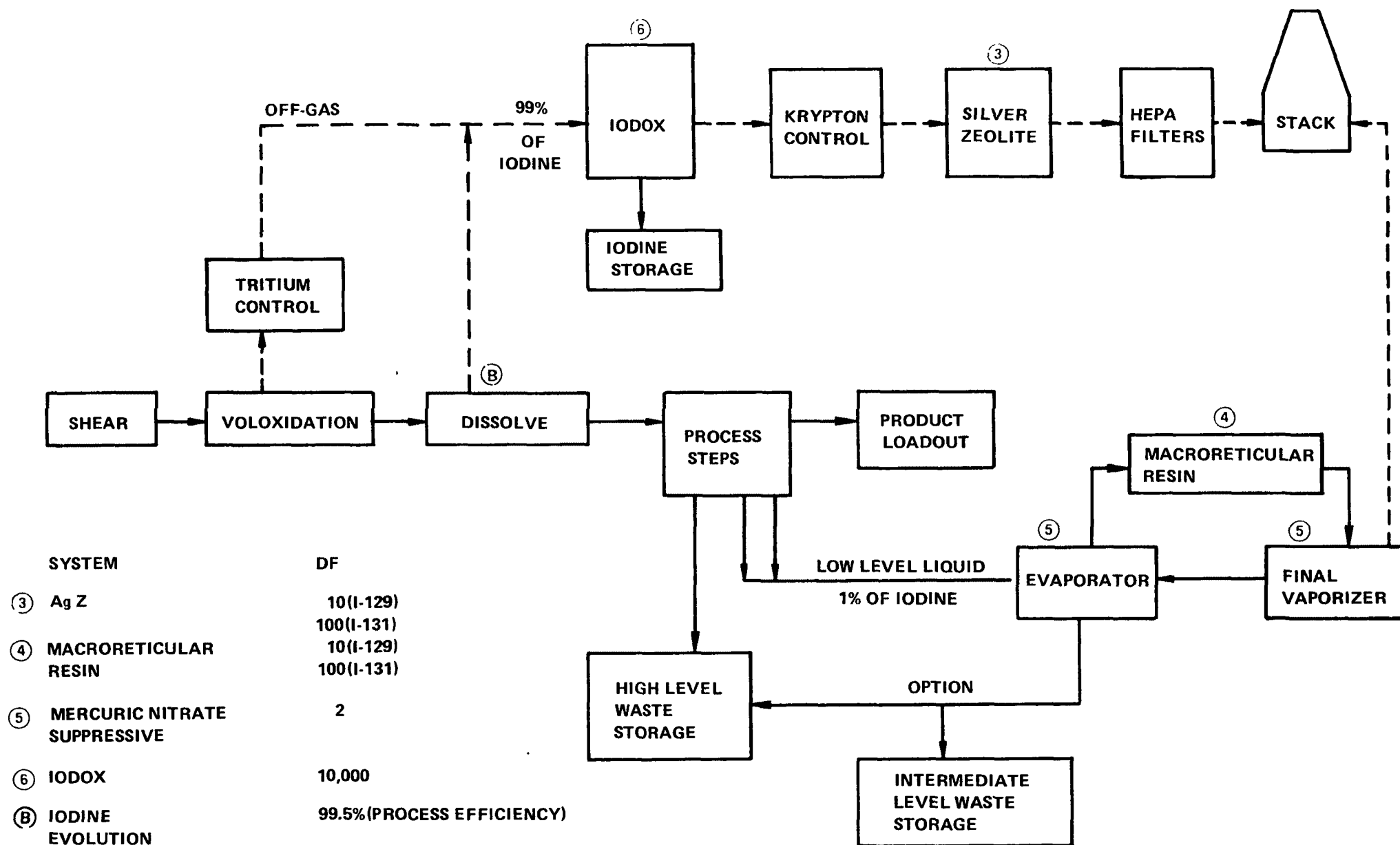
FIGURE 2. SIMPLIFIED SCHEMATIC OF ADVANCED IODINE CONTROL SYSTEMS AT REPROCESSING PLANTS

Table 1.  Krypton control cost summary. [a]

| Process | DF | Capital Cost (M$) | Annual Operating Cost (M$) | Total Annualized Cost (M$) | Present Worth at 10% and 20 years (M$) |
|---|---|---|---|---|---|
| 1. Cryogenic Distillation | 10-100 | 3.4 | 0.12 | 0.73 | 6.2 |
| 2. Selective Adsorption | 10-100 | 3.9 | 0.40 | 1.1 | 9.4 |

a)  All costs are expressed in millions of 1974 dollars.

Table 2. Iodine control cost summary. [a]

| Process | DF | Capital Cost (M$) | Annual Operating Cost (M$) | Total Annualized Cost (M$) | Present Worth @ 10% & 20 years (M$) |
|---|---|---|---|---|---|
| 1. Caustic Scrubbing | 10 | 0.34 | 0.04 | 0.10 | 0.82 |
| 2. Mercuric Nitrate Scrubbing | 10 | 0.31 | 0.12 | 0.18 | 1.5 |
| 3. Silver Zeolite Beds | 10 (I-129) 10 (I-131) | 0.44 | 0.15 | 0.13 | 2.0 |
| 4. Adsorption on Macroreticular Resins | 10 (I-129) 100 (I-131) | 0.14 | 0.04 | 0.065 | 0.56 |
| 5. Mercurie Nitrate Suppression | 2 | 0.14 | 0.04 | 0.065 | 0.56 |
| 6. Iodox | 10,000 | 2.07 | 0.22 | 0.59 | 5.0 |
| A. Voloxidation [b] | 4 [c] | 2.74 | 0.29 | 0.78 | 6.6 |
| B. Iodine Evolution | 200 [c] | 0.75 | 0.08 | 0.21 | 1.8 |

a) All costs are expressed in millions of 1974 dollars.

b) This system is not installed, primarily, to facilitate iodine control, and is listed only for completeness.

c) These values do not represent actual DF's, but represent a process efficiency factor.

Table 3. Comulative environmental dose commitment and potential health effects attributable to Kr-85 discharges from a 1500 MTHM/year reprocessing plant[a]

| Source Term (Ci/yr) | DF | Exposed Organ | Dose Commitment (person-kilorems) | | Health Effects |
|---|---|---|---|---|---|
| $1.6 \times 10^7$ | 1 | whole body | 150 | | 60 |
| | | lungs | 300 | | 14 |
| | | gonds | 82 | | 25 |
| | | | | total | 99 |
| $1.6 \times 10^6$ | 10 | whole body | 15 | | 6.0 |
| | | lungs | 30 | | 1.4 |
| | | gonads | 82 | | 2.5 |
| | | | | total | 9.9 |
| $1.6 \times 10^5$ | 100 | ·whole body | 1.5 | | 0.60 |
| | | lungs | 3.0 | | 0.14 |
| | | gonads | 0.82 | | 0.25 |
| | | | | total | 0.99 |

a) Dose commitments are displayed for a plant operating life of 20 years beginning in 1983.

Table 4. 100-year cumulative environmental dose commitment and estimated health effects attributable to release of I-129 from a 1500 MTHM/yr reprocessing plant. [a,b]

| Source Term (Ci/yr) | DF | Thyroid Dose Commitment (person-kilorems) | Health Effects |
|---|---|---|---|
| 60 | 1 | 1700 | 100 |
| 6 | 10 | 170 | 10 |
| 1.2 | 50 | 34 | 2 |
| 0.6 | 100 | 17 | 1 |
| 0.3 | 200 | 8.6 | 0.5 |
| .06 | 1000 | 1.7 | 0.1 |

a) Partial environmental dose commitment and health effects are calculated for 100 years following release only and for a plant operating life of 20 years.

b) Doses and health effects do not include short term, local impact of either iodine-129 or iodine-131. These are estimated to be 15 and 35 health effects, respectively, for a DF or 1.

Table 5.  Maximum individual thyroid doses from I-129 discharged from a 1500 MTHM/year
reprocessing plant .

| DF | Source Term (Ci/yr) [a] | I-129 Thyroid Dose (mrem/yr) [b] | | | |
|---|---|---|---|---|---|
| | | 6 month old | 4 year old | 14 year old | adult |
| 1 | 60 | 1100 | 1600 | 600 | 140 |
| 10 | 6 | 110 | 160 | 60 | 14 |
| 50 | 1.2 | 22 | 32 | 12 | 2.8 |
| 100 | 0.6 | 11 | 16 | 6 | 1.4 |
| 200 | 0.3 | 5.5 | 8 | 3 | 0.7 |
| 1000 | 0.06 | 1.1 | 1.6 | 0.6 | 0.14 |

a)  The elemental iodine fraction is assumed to be 50.

b)  Atmospheric dispersion coefficient equals $5 \times 10^8$ seconds per cubic meter; only the milk pathway is considered.

Table 6. Maximum individual doses from I-131 discharged
from a 1500 MTHM/year reprocessing plant.

| DF | Source Term (Ci/yr) [a] | I-129 Thyroid Dose (mrem/yr) [b] | | | |
|---|---|---|---|---|---|
| | | 6 month old | 4 year old | 14 year old | adult |
| 1 | 1400 | 1900 | 2300 | 430 | 110 |
| 10 | 140 | 190 | 230 | 43 | 11 |
| 100 | 14 | 19 | 23 | 4.3 | 1.1 |
| 200 | 7 | 9.5 | 12 | 2.2 | 0.54 |
| 500 | 2.8 | 3.8 | 4.6 | 0.86 | 0.22 |
| 1000 | 1.4 | 1.9 | 2.3 | 0.43 | 0.11 |
| 10000 | 0.14 | 0.19 | 0.23 | 0.043 | 0.011 |

a) Fuel cooled for 160 days before processing; the elemental iodine fraction is assumed to be 50%.

b) Atmospheric dispersion coefficient equal $5 \times 10^8$ seconds per cubic meter; all pathways.are considered.

Table 7. Cost effectiveness of krypton and iodine control system.

| System | Equipment | Cost Increment (M$) | Health Effects Adverted | Cost/Health Effect (M$/HE) |
|---|---|---|---|---|
| 1. Krypton | (a) Cryogenic distillation | 6.2 | 89-98 | 0.063-0.070 |
| | (b) Selective absorption | 9,4 | 89-98 | 0.096-0.106 |
| 2. Iodine (for off-gas without iodine evolution) | (a) Scrubber ($HgNO_3$ ) | 1.5 | 121 | 0.012 |
| | (b) Iodox (no scrubbers) | 5.0 | 135 | 0.037 |
| | (c) Second caustic scrubber | 0.82 | 12 | 0.068 |
| | (d) Silver zeolite (one (one scrubber) | 2.10 | 13 | 0.154 |
| 3. Iodine (for off-gas with iodine evolution) | (a) Scrubber [a] | 3.3 | 135 | 0.024 |
| | (b) Iodox (no scrubbers)[a] | 6.8 | 150 | 0.045 |
| | (c) Second scrubber | 0.82 | 13 | 0.063 |
| | (d) Silver zeolite (one scrubber) | 2.06 | 14 | 0.143 |
| 4. Iodine (for liquid streams without iodine evolution) | (a) Macroreticular resin | 0.56 | 14 | 0.040 |
| | (b) Mercuric nitrate suppression | 0.56 | 0.75 | 0.75 |
| 5. Iodine (for liquid streams with evolution) | (a) Macroreticular resin | 0.56 | 0.68 | 0.82 |
| | (b) Mercuric nitrate suppression | 0.56 | 0.038 | 14.7 |

a) Add incremental iodine evolution cost

SUPPLEMENT   G


TRANSURANIUM EFFLUENTS FROM RE-ENRICHING

OR REFABRICATING REPROCESSED URANIUM

# TRANSURANIUM EFFLUENTS FROM RE-ENRICHING OR REFABRICATING REPROCESSED URANIUM

Uranium feed material, either to an enrichment plant or to a fabrication plant, which has been previously used as fuel in a nuclear power plant may still contain trace amounts of radioactive impurities after decontamination at fuel reprocessing.

Spent reactor fuel is typically allowed to decay either at the reactor plant site or at the chemical reprocessing plant site a minimum decay time of 150 to 180 days. The fuel is then dissolved in nitric acid and processed by solvent extraction

The $UF_6$ product from chemical reprocessing will contain small quantities of fission products and transuranium isotopes. Specifications have been published by the Atomic Energy Commission[1] which indicate the maximum acceptable limits for radioactivity resulting from these impurities. These are: gross alpha due to transuranium isotopes -- 1500 dis/min/(g of U); gross beta due to fission products and transuranium isotopes -- 10% of the beta activity of aged normal uranium; and gross gamma due to fission products and transuranium isotopes -- 20% of the gamma activity of aged normal uranium.

Such processed uranium may then be sent to the enriching plant. The above maximum acceptable limit for gross alpha radioactivity can be translated into the following typical distribution (assuming total solvent extraction plus conversion decontamination factors[2] for

neptunium of $10^3$, plutonium - $10^7$, and transplutonium - $10^9$):

neptunium - 9 X $10^2$ alpha dis/min/(g of U), plutonium - 5 X $10^2$ alpha

dis/min/(g of U) and transplutonium - 1 X $10^2$ alpha dis/min/ (g of U).

The actual alpha activity distribution will depend on reactor type,

fuel irradiation history, type·of chemical process, and the additional

conversion and purification operations used in converting uranyl

nitrate hexahydrate to $UF_6$, but should not vary significantly from these

typical values.

The above beta-gamma radioactivity limits are based on gross

radioactivity measurements related to the background of aged normal

uranium. The beta activity limit is based on direct measurement of the

beta counting ratio, and therefore depends upon the variation of counting

efficiency with energy. The gamma specification is based on a

comparative measurement using aged natural uranium and a high pressure

ion chamber. A reasonable gamma comparison with natural uranium can

therefore be equated to 20% of the gamma power of aged normal uranium.

The gamma power of aged normal uranium can be calculated to be 269 MeV/

sec/(g of U), which results in a gamma specification of approximately

54 MeV/sec/(g of U).

Typical reactor return material has shown the fission product

gamma radioactivity distribution given in Table 1. Technetium and

uranium beta and uranium and transuranium alpha radioactivity levels

found are also indicated.

## TABLE 1

CALCULATED GAMMA RADIOACTIVITY DISTRIBUTION OF FISSION PRODUCTS, GAMMA
AND BETA RADIOACTIVITY OF ALL FISSION PRODUCTS, AND ALPHA RADIOACTIVITY
OF TRANSURANIUM AND URANIUM ISOTOPES[a](2)

| Isotope | % of Gamma | Typical distribution based on gamma specification ($\gamma$ MeV/sec/g U) | Radioactivity (Ci/g U) |
|---|---|---|---|
| | | | $\gamma$ Radioactivity |
| Ru-106 | 75 | 40.0 | $42.2 \times 10^{-10}$ |
| Zr-95-Nb-95 | 22 | 12.0 | $9.3 \times 10^{-10}$ |
| Cs-137 | 1 | 0.054 | $\sim 6.9 \times 10^{-11}$ |
| Ce-144 | 1 | 0.054 | $\sim 6.9 \times 10^{-11}$ |
| Other fission products[b] | 1 | 0.054 | $\sim 6.9 \times 10^{-11}$ |
| | | | $\beta$ Radioactivity |
| Tc-99 | -- | | $3.16 \times 10^{-8}$ |
| U-237 | -- | | $2.41 \times 10^{-6}$ |
| | | | $\alpha$ Radioactivity |
| Transneptunium[c] | -- | -- | $2.43 \times 10^{-10}$ |
| Np-237 | -- | -- | $4.32 \times 10^{-10}$ |
| U-232 | -- | -- | $9.01 \times 10^{-9}$ |
| U-233 | -- | -- | $4.70 \times 10^{-11}$ |
| U-234 | -- | -- | $7.59 \times 10^{-7}$ |
| U-235 | -- | -- | $1.71 \times 10^{-8}$ |
| U-236 | -- | -- | $2.88 \times 10^{-7}$ |
| U-238 | -- | -- | $3.14 \times 10^{-7}$ |

[a]Power reactor returns are based on an initial feed of 3.2% U-235, specific power 30 MW/metric ton uranium, exposure 33,000 MW day/metric ton, decay 180 days.

[b]These fission products consist principally of Sr, Sb, Sn, and Te.

[c]Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Cm-242, Cm-244

These radioactivities can be used to determine the annual inputs and system equilibrium concentrations at an enrichment plant (Table 2). The technetium-99 beta will contribute the remaining beta radioactivity and is also included. Plutonium and neptunium concentrations are based on the above specifications for transuranium isotopes in the reactor return material.

Gaseous diffusion operating experience, although of almost 30 years duration, has been very limited in terms of large throughputs of power reactor returns. Although there has been considerable production reactor material returned to the cascade, irradiation exposure of that material has been ten- to twenty-fold less than that for power reactors. Experience to date has indicated the following:[2]

1. A significant quantity of all non-uranium radioactivity (neptunium, plutonium, and fission products) is retained in the feed cylinder ($UF_6$ tank) and will be removed when and where the returned cylinder is washed.

2. $PuF_6$ and $NpF_6$ are easily reduced and therefore removed by trapping with $CoF_2$, $MgF_2$, NaF, Cryolite, etc.

3. Fission product removal (except technetium) by these traps may also be significant. However, good data based on low-level radioactivity feed materials have not been obtained.

4. Technetium, compared to other fission or alpha emission products, is less likely to be removed by any process. Experience at ORGDP* indicates that technetium release to the environment would be 10% of feed to the liquid effluent and 1% of feed to the gaseous effluent.

*Oak Ridge Gaseous Diffusion Plant

TABLE 2

CALCULATED FISSION PRODUCT AND TRANSURANIUM ISOTOPE[a]
ANNUAL INPUTS AND EQUILIBRIUM SYSTEM[e] CONCENTRATIONS[2]

| Isotope | Annual Input (Ci/year) | Equilibrium System burden (Ci) |
|---|---|---|
| Ru-106 | 9.3 | 13.5 |
| Zr-95-Nb-95 | 2.0 | 0.5 |
| Cs-137 | 0.16 | $\dfrac{0.16\ (1-e^{-0.0266T})}{0.0266}$ [b] |
| Ce-144 | 0.16 | 0.17 |
| Other fission products | 0.16 | $0.7^{c}$ |
| Tc-99 ($\beta$ only) | 70.0 | $70.0T^{d}$ |
| Np-237 | 0.9 | $0.9T^{d}$ |
| Transneptunium | 0.5 | $0.5T^{d}$ |

[a]Based on fuel specifications of Table 1.

[b]Not an equilibrium condition since Cs-137 has a 26-year half-life and true equilibrium would only be approached in 130 years. Therefore, activity depends on time, T (years of operation).

[c]Assuming an average effective half-life of 3 years.

[d]Very long half-life, never reaches equilibrium.

[e]8.75 MSWU

5. Experience also indicates that other fission products and alpha radioactivity release fractions should be no more than one tenth of that for technetium. Measurements of gaseous and liquid effluents have failed to identify any other fission products. However release fractions of 1% to the liquid effluent and 0.1% to the gaseous effluent for other fission products will be used below to estimate environmental releases.

6. Cobaltous fluoride traps exhibit decontamination factors of 400 for neptunium and $10^5$ for plutonium prior to feeding to the cascade or conversion facility. Releases for the system after trapping can then be proportioned to those exhibited for uranium in ORGDP release data. Thus, alpha release fractions will be 4 X $10^{-6}$ to the liquid and 2 X $10^{-7}$ to the gaseous effluents for neptunium and 1.6 X $10^{-8}$ to the liquid and 8.0 X $10^{-10}$ to the gaseous effluents for plutonium.

7. A large portion of the radioactivity entering a settling pond will be entrained in the sludge of the pond.

Releases to the environment can occur in three physical states (gas, liquid, and solid). The bulk of the radioactivity will be released as solids, either entrained on adsorbate or equipment removed from service for disposal. Liquid waste will be generated by rinsing (decontamination) of recycled equipment. The first rinse solution, which contains the bulk of the radioactivity, are saved to be used as the dilute acid wash solution. Subsequent rinses are sent to the primary holding pond.

Gaseous wastes can result from purge system venting, venting of evaporator overheads at the uranium recovery facility, and venting of decontamination hoods in the recycle facility. However, the exact breakdown for retention and release factors for each step is not known. One can only make assumptions based on experience with gaseous diffusion. The limited experience available was used to arrive at the following estimates (see Table 3) about gaseous, liquid, and solid discharges for non-uranium radioactivity.[2]

TABLE 3

ASSUMED DISTRIBUTION OF FISSION PRODUCTS AND TRANSURANIUM ISOTOPES
TO ATMOSPHERE, PRIMARY HOLDING POND, AND BURIAL GROUND

| Isotope | Fraction released to atmosphere | Fraction released to primary holding pond | Fraction input to burial ground |
|---|---|---|---|
| Np-237 | $2 \times 10^{-7}$ | $4 \times 10^{-6}$ | ∿1.0 |
| Other Transuranium | $8 \times 10^{-10}$ | $1.6 \times 10^{-8}$ | ∿1.0 |
| Tc-99 | 0.01 | 0.10 | 0.89 |
| Fission Products | 0.001 | 0.01 | 0.989 |

Primary enrichment plant sources of gaseous radioactive wastes are the product and waste purge systems. Uranium particulates are removed from these process streams by the high-efficiency-particulate absolute (HEPA) filter, which has an efficiency greater than 99.95%. Removal of gaseous uranium is achieved through the use of two chemical traps in the product and waste withdrawal systems, in series, between the cold trap and point of discharge into the air.

The first trap contains sodium fluoride that provides for the adsorption of uranium and certain fission or alpha emitting products. Through heating and proper valving, the trapped uranium may be desorbed and subsequently returned to the cascade. The second trap in the series contains alumina that is used for further removal of uranium prior to discharbe of the gas stream to the atmosphere. This trap is nonreversible and uranium recovery is accomplished by leaching with nitric acid.

The fraction of the feed made up of reactor returns is passed through cobaltous fluoride traps prior to being fed into the cascade[2]; the traps remove plutonium, neptumium, and a major fraction of the fission products. These products are removed from the gas stream by·reduction with $CoF_2$ to the tetraflouride forms that, being particulates, are entrained within the traps.

Quantification of potential gaseous effluents is difficult because of uncertainties about the behavior of certain fission products in feed cylinders, traps, piping, and equipment. In attempting to analyze

possible releases to the environment, all assumptions, where necessary, have been made so as to overestimate the magnitude of the source term. Uranium and technetium releases were estimated by comparison with operating experience and extrapolated to higher operating levels. Fission product releases were based on current fission product specifications, with releases being assumed proportional to that of technetium, with the exception that a decontamination factor (DF) and/or retention factor 10 times that for technetium was assumed. This assumption is very conservative, since current experimental investigations indicate that the actual factor might be as high as 100 to 1000.[2] Releases of the alpha emitters, neptunium and plutonium, were estimated by assuming an alpha specification of 1500 dis/min/(g of U) in reactor returns, with a neptunium DF of 400 and a plutonium DF of $10^5$ through $CoF_2$ traps. Once fed into the cascade, neptunium and plutonium are assumed to be released to the environment in the same proportions as uranium.

The estimated constituents of an effluent under the above assumptions are listed in Table 4.

It may be concluded that recycled uranium which has been re-enriched will present no particular problem at the fabrication plant because most of the impurities of higher isotopes have been taken out in the enriching process, and could not make a significant contribution to an industry limit of 0.5 mCi/GW(e) for alpha-emitting transuranics of half-life greater than one year.

## TABLE 4

ESTIMATED RADIOACTIVITY RELEASED TO THE ATMOSPHERE FROM
AN ENRICHMENT PLANT[d]
(Transuranic alpha specification = 1,500 dis/min/g U)

| Isotope | Radioactivity (Ci/year)/Gw(e) |
|---|---|
| U-232 | $2.75 \times 10^{-8}$ |
| U-233 | $1.5 \times 10^{-10}$ |
| U-234 | $3.25 \times 10^{-5}$ |
| U-235 | $1.25 \times 10^{-6}$ |
| U-236 | $0.92 \times 10^{-6}$ |
| U-238 | $5.3 \times 10^{-6}$ |
| Transneptunium[b] | $3.3 \times 10^{-13}$ |
| Np-237[c] | $1.7 \times 10^{-10}$ . |
| Tc-99 | $4.5 \times 10^{-4}$ |
| Ru-106 | $6.0 \times 10^{-6}$ |
| Zr-95-Nb-95 | $1.25 \times 10^{-6}$ |
| Cs-137 | $0.92 \times 10^{-7}$ |
| Ce-144 | $0.92 \times 10^{-7}$ |
| Other fission products | $0.92 \times 10^{-7}$ |

[a]Relative to Tc-99, the retention of all fission
products in equipment or traps is greater by a factor of 10.

[b]Cobaltous fluoride trap decontamination factor for
Pu-239 = $10^5$.

[c]Cobaltous fluoride trap decontamination factor for
Np-237 = 400.

[d]8.75 MSWU Plant

If, however, recycled material goes directly from reprocessing to fabrication, cleanup systems will have to be designed and installed to collect the impurities as the material is converted from $UF_6$ to $UO_2$ for blending and/or pelletizing. These systems should have efficiencies and decontamination factors similar to those described above for the enrichment plant. They would, therefore, be expected to also reduce transuranium isotopes in the $UO_2$ to levels resulting in negligible releases compared to the proposed standard of 0.5 mCi/GW(e).

## REFERENCES

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(2) "Environmental Statement - Expansion of U.S. Uranium Enrichment Capacity," U.S. Energy Research and Development Administration, DRAFT ERDA-1543, June 1975.

SUPPLEMENT    H


ENVIRONMENTAL ANALYSIS OF THE URANIUM FUEL CYCLE,

PART I (FUEL SUPPLY):   URANIUM MILLING - REVISED

CONTENTS

TABLES

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CONTENTS (CONTINUED)

## 1.0 Introduction

The EPA recently completed a technical review (1) of the uranium milling industry as part of an overall analysis of the uranium fuel cycle (2) (3). This review included a description of the milling process, estimations of radioactive effluent releases, radiological impact, health effects impact, and the costs and effectiveness of control technologies for mills. An analysis of the tailings piles associated with mills was also included. This review was prepared in support of EPA's proposed standards for the nuclear fuel cycle, 40 CFR Part 190 (4).

Since publication in 1973, considerable new information on the uranium milling industry has become available (5,6,7,8); in particular, the engineering survey report (6), "Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing 'as Low as Practicable' Guides - Milling of Uranium Ores," has been prepared by Oak Ridge National Laboratory for the Nuclear Regulatory Commission (NRC). This report contains an extensive review of the costs and the effectiveness of various control technology systems for uranium mills and mill tailings piles.

The EPA believes it to be worthwhile to revise its previous technical review of the milling industry, taking into account these new sources of information. Because radon-222 releases from fuel cycle facilities have been specifically excluded from EPA's proposed standard, analysis of radon-222 releases from uranium mills and uranium mill tailings piles has been omitted from this document. Radon-222 will be the subject of separate regulatory actions at a later date.

## 2.0 General Description of the Milling Process

A uranium mill extracts uranium from ore. The product is a semi-refined uranium compound ($U_3O_8$) called "yellowcake" which is the feed material for the production of uranium hexafluoride ($UF_6$). As of March 1975, seventeen mills (7) were operating in the United States (table 2.0-1) with nominal capacities ranging from 250 to 7,000 tons of ore per day. These mills are characteristically located in arid, isolated regions of the west. Areas with significant high grade ore reserves are (6): Wyoming, 55 million tons; New Mexico, 50 million tons; Texas, 11 million tons; Colorado - Utah, 6 million tons; all other areas combined, 7 million tons.

Eighty percent of yellowcake is currently produced by a process that uses sulfuric acid to leach the uranium out of the ore; the remainder is produced by a sodium carbonate, alkali leach process. Exact details vary from mill to mill, but, as an example, the principal steps in an acid leach process mill are as follows:

a. Ore is blended and crushed to pass through a 2.5 cm (1 inch) screen. The crushed ore is then wet ground in a rod or ball mill and is transferred as a slurry to leaching tanks.

b. The ore is contacted with sulfuric acid solution and an oxidizing reagent to leach uranium from the ore. The product liquor is pumped to the solvent-extraction circuit while the washed residues (tailings) are sent to the tailings pond or pile.

c. Solvent extraction or ion exchange is used to purify and concentrate the uranium.

Table 2.0-1 (7)

URANIUM MILLS IN OPERATION AS OF MARCH 1975

| COMPANY | LOCATION | YEAR OPERATIONS INITIATED | NOMINAL CAPACITY (Tons of Ore/Day) |
|---------|----------|---------------------------|-------------------------------------|
| Anaconda Company | Grants, New Mexico | 1953 | 3000 |
| Atlas Corporation | Moab, Urah | 1956 | 800-1500 |
| Conoco & Pioneer Nuclear, Inc. | Falls City, Texas | 1961 | 220-1750 |
| Cotter Corporation | Canon City, Colorado | 1958 | 150-450 |
| Dawn Mining Company | Ford, Washington | 1957 | 0-400 |
| Exxon, U.S.A. | Powder River Basin, Wyoming | 1971 | 2000 |
| Federal-American Partners | Gas Hills, Wyoming | 1959 | 500-950 |
| Kerr-McGee Nuclear | Grants, New Mexico | 1958 | 3600-7000 |
| Petrotomics Company | Shirley Basin, Wyoming | 1962 | 525-1500 |
| Rio Algom Corp. | La Sal, Utah | 1972 | 500 |
| Union Carbide Corp. | Uravan, Colorado | 1950 | 0-1300 |
| Union Carbide Corp. | Natrona County, Wyoming | 1960 | 1000 |

Table 2.0-1 (Continued)

| COMPANY | LOCATION | YEAR OPERATIONS INITIATED | NOMINAL CAPACITY (Tons of Ore/Day) |
|---------|----------|---------------------------|------------------------------------|
| United Nuclear-Homestake Partners | Grants, New Mexico | 1958 | 1650-3500 |
| Utah International, Inc. | Gas Hills, Wyoming | 1958 | 750-1200 |
| Utah International, Inc. | Shirley Basin, Wyoming | 1971 | 1200 |
| Western Nuclear, Inc. | Jeffrey City, Wyoming | 1957 | 400-1200 |
| TVA (Mines Development, Inc.) | Edgemont, South Dakota | 1956 | 250-500 |

d. The uranium is precipitated with ammonia and transferred as a slurry.

e. Thickening and centifuging are used to separate the uranium concentrate from residual liquids.

f. The concentrate is dried at 400°F and is sometimes calcinated at 750 to 950°F.

g. The concentrate or yellowcake is packaged in 208 liter (55 gallon) drums for shipment.

Large amounts of solid waste tailings remain following the removal of the uranium from the ore. A typical mill may generate 1,800 metric tons per day of tailings solids slurried in 2,500 metric tons of waste milling solutions. Over the lifetime of the mill, 100 to 200 acres may permanently be committed to store this material. These "tailings piles" will have a radiological impact on the environment through the air pathway by continuous discharge of radon-222 gas (a daughter of radium-226), through gamma rays given off by radium-226, radon-222 and daughters as they undergo radioactive decay, and finally through air and water pathways if radium-226 and thorium-230 are blown off the pile by wind or are leached from the pile into surface waters.

## 3.0 Releases of Radioactive Effluent from Uranium Mills

The radioactivity associated with uranium mill effluents comes from the natural uranium and its daughter products present in the ore. During the milling process, the bulk of the natural uranium is separated and concentrated, while most of the radioactive daughter products of uranium remain in the uranium-depleted solid residues that are pumped to the tailings retention system. Liquid and solid wastes from the milling operation will contain low level concentrations of these radioactive materials, and airborne radioactive releases include radon gas and particles of the ore and the product uranium oxide. External gamma radiation levels associated with uranium milling processes are low, rarely exceeding a few mrem/hr even at surfaces of process vessels.

## 3.1 Airborne Releases

Airborne releases from uranium milling operations include both particulate matter and gases. Dusts containing uranium and uranium daughter products (thorium-230 and radium-226) are released from ore piled outside the mill. Dusts containing uranium and uranium daughter products are released from the ore crushing and grinding ventilation system, while a dust containing mostly uranium without daughters is released from the yellowcake drying and packaging operations. These dusts are discharged to the atmosphere by means of low stacks.

Because uranium is discharged to the air pathway as ore dust and as calcinated yellowcake, it will be considered as an insoluble aerosol. Radium-226 and thorium-230 discharged as ore dust will also

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be considered insoluble aerosols.

The air flow through a typical crushing and grinding ventilation system is about 27,000 cfm; that through the yellowcake drying and packaging ventilation system is about 6,000 cfm. Because of the different air flows, dust characteristics, and locations within the plant, separate air cleaning equipment systems are usually required; a mill is therefore usually considered to have two separate airborne effluent release streams, each with its own control systems, costs, and source terms.

Radon gas is released from the leach tank vents, ore piles, tailings retention system, and the ore crushing and grinding ventilation system. There is no practical method presently identifiable that will prevent the release of radon gas from uranium mills.

As an example, table 3.1-1 gives the estimated maximum release rates and conservative estimates of site boundary concentrations considering all potential sources of airborne dust fumes and mists as predicted for the Highland Uranium Mill in Wyoming (9,10). The capacity of the Highland Mill is about 1,200 MT/yr of yellowcake.

Toward the end of the operating lifetime of a tailings retention system, some of the tailings will no longer be under water and will dry out to form a beach (6). Wind erosion can then carry off tailings material as airborne particulate matter unless control measures are taken to prevent such erosion.

Table 3.1-1 (9,10)

Predicted airborne releases of radioactive materials from the Highland Uranium Mill

| Radionuclide | Release rate (Ci/yr) | Site boundary A [a] Air concentration (pCi/m$^3$) | Site boundary B [b] Air concentration (pCi/m$^3$) |
|---|---|---|---|
| Uranium-natural | 0.1 | 0.003 | 0.0004 |
| Thorium-230 (insoluble) | .06 | .001 | .0001 |
| Radium-226 (insoluble) | .06 | .001 | .0001 |

[a]Distance to site boundary A assumed to be 800 m (2,600 ft) west of mill.

[b]Distance to site boundary B assumed to be 5,200 m (12,700 ft) east of mill.

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## 3.2 Waterborne Releases

The following discussion refers to the best of current procedures of handling mill liquid wastes, in which these wastes plus tailings are stored in a tailings retention pond system which uses an impervious clay-cored earth dam combined with local topographic features of the area to form an impoundment.

The liquid effluent from an acid-leach process mill consists of waste solutions from the leaching, grinding, extraction and washing circuits of the mill. These solutions, which have an initial pH of 1.5 to 2, contain the unreacted portion of the sulfuric acid used as the leaching agent in the mill process, sulfates, and some silica as the primary dissolved solids, along with trace quantities of soluble metals and organic solvents. This liquid is discharged with the solids into the tailings pond.

Concentrations of radioactive materials predicted in the 2,500 MT/day of waste liquor from the Highland milling plant are shown in table 3.2-1 (9,10). Radioactive products of radon decay may also be present in small concentrations. Since the concentrations of radium-226 and thorium-230 are about an order of magnitude above the specified limits to 10 CFR 20, considerable effort must be exerted to prevent any release of this material from the site. The waste liquor is, therefore, stored in the tailings retention pond which is constructed to prevent discharge into the surface water system and to minimize percolation into the ground. This is a continuing potential problem requiring monitoring programs to insure that there is no significant movement of contaminated liquids into the environment.

Table 3.2-1

Concentrations of radioactive effluents in
waste liquor from the Highland uranium mill (9,10)

| Radionuclide | Concentration (pCi/l) |
|---|---|
| Uranium-natural | 800[a] |
| Radium-226 | 350 |
| Thorium-230 | 22,000 |

[a]About 0.001 g/ml.

If an earth-fill, clay-cored dam retention system serves as
a collection and storage system for the liquid and solid process
wastes generated in the mill, it will permit the evaporation of most
of the contained waste liquids and serve as a permanent receptacle
for the residual solid tailings.  However, after the initial
construction of the retention system, it is to be expected that
there will be some seepage of radionuclides through and around the
dam (9,10) and downward into the soil beneath the impoundment area.
It has been estimated that this seepage will diminish over a period
of about 2 years because of the sealing effect from accumulation
of finer particles between the sandstone grains.  On the other hand,
sealing may not occur.  Examples of the total quantities of radionuclides
that are estimated to be released through and around the dam are shown
in table 3.2-2.  Radium-226 is a radionuclide of concern in this case.
Radium-226 levels as high as 32 pCi/l (11) have been found in seepage
from current operating mills.  Assuming a seepage rate of 300 liters
per minute, the concentration of radium-226 seeping into a stream of
140 liters per second (5 cubic feet per second) is approximately 1 pCi/l
which is 1/5 of EPA's proposed interim Primary Drinking Water Regulation
for radium-226 (12).  In the applicant's environmental report for the
Highland Uranium Mill (9,10), a seepage concentration of 350 pCi/l
radium-226 was assumed, bringing the concentration of radium in such an
offsite stream up to 12 pCi/l.  The Highland Uranium Mill is also esti-
mated to release to the tailings pond 22,000 pCi/l thorium-230 and trace
quantities of short-lived radon daughter products.

Table 3.2-2

Estimates of quantities of radionuclides seeping through the

impoundment dam of a uranium mill initially and at 2-1/4 years (9,10)

| Radionuclide | Initial seepage per day | Seepage per day[a] after 2-1/4 years |
|---|---|---|
| Uranium | 350 μCi | 35 μCi to 3.5 μCi |
| Thorium-230 | 9,600 μCi | 960 μCi to 96 μCi |
| Radium-226 | 150 μCi | 15 μCi to 1.5 μCi |

[a]Seepage assumed to be inhibited due to sealings effect from accumulation of fines between sandstone grains.

As an additional example, the analysis of plant tailings
effluents for the Humeca Uranium Mill, which uses an alkaline lead
process, is given in table 3.2-3 (13).

The radiological significance of seepage from tailings ponds
will depend on the location of the pond.  In arid regions, the
seepage may evaporate before leaving the site, leaving the radio-
activity entrained and absorbed on soil.  Should the tailings pond
be located near a river, minor leakage might be diluted sufficiently
by the additional river water to meet relevant drinking water standards.
Discharge of pond seepage into streams providing insufficient dilution
and not under the control of the licensee would not be acceptable.  In
such cases, a secondary dam may be built below the primary dam to
catch the seepage which may then be pumped back into the tailings ponds.

Table 3.2-3 (**13**)

Analysis of plant tailings effluents

from the Humeca Uranium Mill

(alkaline leach process)

| Radionuclide | pCi/l |
|--------------|-------|
| Radium-226 | 10 to 2,000 |
| Thorium-230 | 0.1 |
| Uranium-238 | 4,000 |

## 4.0 The Model Uranium Mill

A model plant has been assumed in order to achieve a common base for the comparison of radiation doses, committed health effects, and radioactive effluent control technology.

The model mill is defined in terms of contribution to the nuclear fuel cycle that is consistent with current designing and projected commercial industry practice (6). However, it is not necessarily representative of presently operating facilities. Characteristics of the model mill are assumed to be:

    a.  600,000 MT ore milled per year,

    b.  1,140 MT $U_3O_8$ as yellowcake produced per year,

    c.  use of the acid leach process,

    d.  a tailings retention pond system which uses a clay-core earth dam and local topographic features of the area to form the impoundment,

    e.  collection and return of any seepage through the dam to the tailings pond, and

    f.  location in a western State in an arid, low-populated density region.

While Reference 1 considered the radiological impact of seepage through a model clay core impoundment dam, it is now believed to be standard practice (6) to collect and return any such seepage to the tailings pond so that there are no routine liquid discharges of radio-nuclides to water pathways from mills. The cost of a seepage control

system is nominal compared to the cost of the tailings impoundment system itself.

Radiation dose rates and health effects that might result from the discharges of airborne radioactive effluents from the model mill were calculated using standard $\chi/Q$ values, dose conversion factors, model pathways, and health effect conversion factors that are similar to those for other facilities in the previous discussion of the fuel supply cycle. These factors and assumptions are discussed in Appendix A of Reference 1.

The operating lifetime of a uranium mill is commonly from 12 to 15 years, depending upon the local ore supply and the demand for uranium. In a few instances, the operating lifetime may be longer, and allowances are sometimes made for that possibility if it appears feasible. For the model mill, an operating lifetime of 20 years has been selected.

## 5.0 Radioactive Effluents from a Model Uranium Mill

Because regulations have not required uranium mills to report the total amounts of each radionuclide discharged per year, the source terms chosen for model mills are based on somewhat limited operational information (6). Source terms listed in table 5.0-1 for model mills are believed, however, to be reasonably accurate estimates of the quantities of radioactive materials discharged to air pathways with base case controls. The controls assumed as the base case consist of an orifice scrubber on the crusher and fine ore bins, and a wet impingement scrubber in the yellowcake drying and packaging areas. The milling procedures are so similar for acid and alkaline leach processes that source terms for the two types of mills are considered identical, except that the alkaline leach process does not remove thorium from the ore so that, in this case, there is very little thorium-230 as an impurity in the yellowcake dust.

The model mill is also assumed to use clay-core dam impoundment technology for tailings with a catch basin if required to contain seepage through the dam. Unless the impoundment area is lined with an impervious material, considerable quantities (as much as 10 percent) of the liquid effluent from the mill will leak out through the bottom of the pond. However, because of the ion-exchange properties of most soils, radionuclides dissolved in this effluent will attach to soil particles and will not reach offsite locations or ground water. The model mill is considered, therefore, to deliver no radiation exposure to members of the general population through liquid pathways.

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Table 5.0-1

Discharge of Radionuclides to the Air from Model Uranium Mills[a] and Tailings Piles (6)

With Base Case Controls

| Radionuclide | Chemical or Physical State | Acid Leach Mill Source Term (mCi/yr) | Alkaline Leach Mill Source Term (mCi/yr) |
|---|---|---|---|
| Uranium-238 and 234 | ore dust (oxides) | 9.0 | 9.0 |
| Radium-226 | ore dust | 4.5 | 4.5 |
| Thorium-230 | ore dust | 4.5 | 4.5 |
| Uranium-238 and 234 | yellow cake (oxides) | 170. | 170. |
| Radium-226 | yellow cake | 0.2 | 1.7 |
| Thorium-230 | yellow cake | 4.7 | --- |
| Uranium-238 and 234 | tailings sand (0-10 μ) | 0.2 - 0.8 | 0.3 - 2.2 |
| Radium-226 | tailings sand (0-10 μ) | 1.3 - 4.2 | 2.3 - 1.5 |
| Thorium-230 | tailings sand (0-10 μ) | 1.4 - 4.5 | 2.4 - 1.5 |

[a] 6% moisture ore, radon-222 releases excluded

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Each site must be evaluated individually. If the ground water table is high and the soil is low in ion exchange capacity so that it becomes likely that radium-226 and thorium-230 will escape from the tailings impoundment into underground waters, then the pond area could be lined with an impervious membrane of asphalt to minimize seepage. Acid wastes would have to be neutralized beforehand to prevent damage to this type of liner.

The amount of radioactive particulate material removed from the tailings beach by wind erosion is believed to depend on the area of the beach, the wind velocity, and particle size distribution of the tailings (6). Estimates of this source term are included in table 5.0-1. Particles greater than 10μ in diameter are not considered to be respirable particles and are not included in the inhalation source term pathway. Historically, windblown tailings have caused elevated gamma exposure levels around piles, however, the inhalation pathway has been determined to be the critical pathway. Levels of control sufficient to limit radiation exposure through the inhalation pathway will also prevent, to a significantly greater degree, exposures through the ground deposition, whole body exposure pathway.

## 6.0 Radiological Impact of a Model Mill

Estimates of the radiation doses to individuals through the air pathway in the vicinity of an acid leach model mill using base case controls from routine emissions are shown in table 6.0-1. The estimated collective lung doses to the population in the vicinity of an acid leach mill are given in table 6.0-2. The collective lung dose is determined by summing the average individual radiation dose equivalent to individuals living within 80 kilometers of the mill over the total population within 80 kilometers of the mill. The models for the dispersion and dose calculations are discussed in detail in Appendix A of Reference (1). Based on the information available at the time that analysis was performed, an effective half-life of 1,000 days was used for insoluble class Y compounds in the pulmonary region of the lung in calculating the lung doses from mill emissions. In accordance with what is now becoming accepted practice, in this report all dose conversion factors are calculated using a 500-day effective half-life (18), and are, therefore, reduced by a factor of two from the previously used values.

The dose conversion factor used to calculate the lung dose is believed to be an order of magnitude more conservative than the dose conversion factor used in Reference (6). Reasons for this difference which relate to assumptions regarding lung model parameters, are discussed elsewhere. It is also assumed that food consumed by individuals living near the mill is not produced locally so that exposure through

Table 6.0-1


Radiation Doses to Individuals due to Inhalation
in the Vicinity of a Model Mill with Base Case Controls

| Radionuclide | Source Term (mCi/yr) | Critical Organ | Dose Equivalent to Critical Organ | |
| | | | Individual at Plant Boundary (mrem/yr) | Average Individual Within 80 kms (mrem/yr) |
| --- | --- | --- | --- | --- |
| Uranium-234 and 238 | 180 | Lung | 170 | $3.9 \times 10^{-2}$ |
| Thorium-230 | 15 | Lung | 15 | $3.4 \times 10^{-3}$ |
| Radium-226 | 10 | Lung | 15 | $2.2 \times 10^{-3}$ |
| Total | 205 | | 200 | $4.5 \times 10^{-2}$ |

Table 6.0-2


Collective Dose to the General Population in the
Vicinity of a Model Mill with Base Case Controls

| Radionuclide | Source[a] Term (mCi/yr) | Pathway | Critical Organ | Collective Critical Organ Dose (person rem/yr) |
|---|---|---|---|---|
| Uranium-234 and 238 | 180 | Air | Lung | 2.2 |
| Thorium-230 | 15 | Air | Lung | 0.2 |
| Radium-226 | 10 | Air | Lung | 0.1 |
| | | | Total | 2.5 |

[a]Releases to water pathways assumed equal to zero, and doses from radon-222 are not included.

food chains is not significant compared to lung exposures resulting

from the direct inhallation of radioactive particulate matter. The

radon exposure pathway was excluded from this report.

Because there are no liquid releases from the model mill, there

is no projected radiological impact through water pathways.

## 7.0  Health Effects Impact of a Model Mill

Potential health effects to members of the general population

in the vicinity of a model mill using base case controls are esti-

mated to be 0.0002 lung cancers per year of operation, or 0.005

such effects for 30 years of operation. The models used for the cal-

culation of health effects are given in Appendix A of reference (1).

## 8.0 Control Technology for Uranium Milling

### 8.1 Airborne Effluent Control Technology

Hazardous airborne gaseous and particulate wastes are generated in the milling operation from a number of different sources. The major areas of the milling operations in which gaseous and particulate matter effluents must be controlled are the ore crushing area, the fine ore bins, and the yellowcake drying and packaging areas. Mills often prefer to use multiple dust collection systems rather than design a single, more elaborate system. There will usually be two or more ore dust collectors and separate systems for the yellowcake dryer and for the yellowcake packaging rooms.

Dust collector systems that are currently used or that can be adapted for use by uranium mills are discussed in reference (6). They are for the most part control technologies that have been proven and are standard industrial equipment.

Briefly, these treatment methods are:

a. Orifice Scrubbers - The dusty air flows through a stationary baffle system coated with a sheet of water. The dust particles penetrate the water film and are captured.

b. Wet Impingement Scrubber - The dusty air carrying water droplets added by preconditioning sprays passes through perforated plates to atomize the water and to wet the dust. Particles are then collected by impingement on baffle plates and a vaned demister.

c. Venturi Scrubber - The dusty air is passed through a venturi, increasing its velocity. Water is added which atomizes in the gas stream and collects the dust by impingement. The wetted dust is

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removed by demisters. Raising the pressure drop across the venturi increases the collection efficiency, but this requires higher energy levels and raises the costs.

d. Bag Filters – These filters are made of woven or felted fabric and have high collection efficiencies provided the air being filtered is cool and dry.

e. HEPA Filters – These filters are made of fiber glass. They have very high efficiencies but have a number of limitations; in particular, they can only be used in conjunction with a prefilter and on dry air streams.

Current practice involves the use of wet dust control systems, although several mills use bag filters for air flows from ore handling and from the yellowcake packaging area. The costs and percent effluent reduction for the various control systems suitable for effluent streams of the model mill are given in table 8.1-1.

Particulate material can be prevented from being windblown off the tailings pile beach by back filling with overburden and, as an interim measure, by chemical stabilization by spraying with petroleum derivatives. Chemical stabilization lasts about a year and must be repeated on a regular schedule.

Other sources of gas and dust which can be controlled are the open pit mine haul roads and the ore storage and blending piles. In some instances, the liquid content of the ore as mined may be sufficiently high to eliminate most dust formation in the ore storage and blending area; due to insufficient information, this case

Table 8.1-1

Cost and Efficiencies of Control Technology for Mills[a]

| Control Method | Capital Cost (dollars) | Annual Operating Costs (dollars) | Present Worth[b] (dollars) | Percent Effluent Reduction (%) |
|---|---|---|---|---|
| A. Gaseous (Crusher and Fine Ore Bins) | | | | |
| 1. Orifice Scrubber | 101,000 | 7,200 | 172,000 | 93.6 |
| 2. Wet Impingement Scrubber | 116,000 | 8,600 | 200,000 | 97.9 |
| 3. Low Energy Venturi Scrubber | 173,000 | 17,000 | 340,000 | 99.5 |
| 4. Bag Filters | 300,000 | 21,000 | 506,000 | 99.9 |
| B. Gaseous (Yellowcake Drying and Packaging) | | | | |
| 1. Wet Impingement Scrubber[c] | (35,000) | (3,500) | (69,000) | 97.9 |
| 2. Low Energy Venturi Scrubber[c] | (35,000) | (6,900) | (103,000) | 99.5 |
| 3. High Energy Venturi Scrubber | 46,000 | 15,000 | 193,000 | 99.9 |
| 4. High Energy Venturi Scrubber + HEPA Filters | 106,000 | 22,000 | 322,000 | >99.99 |
| C. Liquids, Solids, and Windblown Particulate Matter | | | | |
| 1. Clay Core Dam Retention System with Seepage Return and 0.6 Meters (2 feet) of Earth Cover Plus Rock Stabilization[e] | 2,250,000 | 50,000[d] | 2,750,000 | |
| 2. Chemical Control of Windblown Dust from Tailings Pond Beach | 63,000 | 8,000 | 142,000 | 100.00 |
| 3. Asphalt Liner for Tailings Pond[e] | 800,000 | 0 | 800,000 | 100.00 |

[a]1974 dollars, radon-222 emissions not included.
[b]Present Worth = Capital Cost + (Annual Cost x 9.818); 8% Discount Rate, 20 yr. Plant Lifetime.
[c]Costs for all yellowcake effluent control are shown for completeness. In actual practice, the value of recovered product more than compensates the cost of control options B1 and B2.
[d]Includes investment to provide for perpetual care.
[e]160 acre tailings pile.

will not be considered at present beyond stating that the problem appears potentially significant and, that it can be controlled in principle through sprinkling and by use of wind breaks. Dust generation on ore haul roads can also be controlled by sprinkling.

## 8.2 Waterborne Effluent Control Technology and Solid Waste Control Technology

New mills in the Rocky Mountains area are using impoundment technology in order to approach zero liquid discharge levels. Recent practice for treatment of solid and liquid wastes is to select a natural ravine which has three basic qualifications for waste storage: (a) limited runoff, (b) dammable downstream openings, and (c) an underlying impermeable geologic formation. Diversion systems (dams and canals) are used to limit the runoff area emptying into the storage basin to prevent flooding of the ravine during a postulated 50-100 year maximum rainfall occurrence. The tailings dam, which should be clay-cored, is keyed into the underlying impermeable formation, which, in one example, is a low porosity shale. Tailings solids slurried in waste process liquids are pumped to the impoundment reservoir for storage and liquid reduction. Liquid reduction is accomplished primarily by evaporation, but also by seepage through the dam, the reservoir walls and floor. By filling a dammed natural depression with tailings, a relatively flat, stable contour is achieved.

Two methods for seepage collection and return are being considered for new mills. Seepage has been estimated to occur from a clay-core retention dam at a rate of 300 liters per minute. In that situation when an impermeable geological formation underlies the retention system, seepage can be collected in a catch basin located at the foot of the dam. The collected seepage can be pumped

back into the retention pond thus eliminating release to the offsite environment. In that situation where either an underlying impermeable geological formation is not existent or is not continuous, vertical seepage may occur to the underlying ground water formation. Wells may be drilled downstream of the retention system into the subsurface formations where seepage will collect, and this water is pumped back to the retention system. Such a system requires specific favorable subsurface conditions. In both cases, these control costs are small compared to the cost of the clay core dam retention system (1).

Impoundment of solids is being accomplished in older mills merely by construction of a dike with natural materials and filling the diked area with slurried tailings. When full, the height of the dike is increased with dried tailings to accommodate even more waste material. Process liquids which overflow the tailings dike or seep through the dike are sometimes routed through a treatment system and discharged to the environment. The diking procedure which is less costly initially, creates an above-ground pile of tailings which is difficult and costly to stabilize. While the mill is operating, this type of pile is also subject to wind and water erosion. Field studies at tailings piles after mill shut-down have shown high gamma radiation levels in the vicinity of such piles, elevated radium-226 levels in water supplies, and high airborne levels of thorium-230 and radium-226 due to wind blown tailings (14,15,16,17). For these reasons, new mills are not likely to be built using this type of solid waste control.

Stabilization of tailings piles requires grading of the tailings area to lessen side slopes, establishing drainage diversion, covering with nonradioactive material, and revegetating the area. In semiarid regions it may be necessary to initially irrigate the pile to achieve vegetation growth. Other types of stabilization may also be feasible. One method involves the covering of the tailings with large aggregate gravel from a river bottom. Silt fines which accompany the river gravel will blow away in a short time leaving what is effectively a wind-proof rip rap, thus significantly reducing or eliminating migration of the tailings outside the controlled area. The costs of such stabilization has recently been estimated (6) at $350/acre-ft for earth, and $2,000/acre-ft for rock. The cost associated with stabilizing a diked surface pile is significantly higher and probably less effective because of difficulties faced in grading, covering, and revegetating the potentially steep side slopes.

Uranium mill tailings piles are long half-life, low-level radioactive wastes. As such, they will require perpetual care. This will include occasional inspection and maintenance to insure integrity of the stabilizing cover, fencing, and of the warning signs around the pile. An annuity should be included as part of the cost of the control technology to pay for this care. The maintenance associated with perpetual care of a stabilized dike system would probably be higher than that for the depression fill system, since there is tendency toward collapse of side slopes and possibly inadequate drainage of precipition from the pile.

## 9.0 Effluent Control Technology for the Model Mill

Typical current effluent control systems were assumed for the model mill. They were:

a. Ore Crusher and Ore Bin Dust - Orifice Scrubber.

b. Yellowcake Dryer and Packaging Dust - Wet Impingement Scrubber.

c. Liquid and Solid Waste - Clay-core dam retention system (160 acres) with seepage return and exposed beach. To be stabilized with 2 feet of earth cover and 6 inches of rock cover.

The radiological impact of total airborne effluent versus successively more effective control systems for a model uranium mill are listed in table 9.0-1. Each improvement in control is the most cost-effective available at that level of control.

The output of the model plant using base case controls is 1,140 MT $U_3O_8$ of which approximately 1% is recovered by the wet impingement dust collector system during drying and packaging operations (6). The value of 11,00 kilograms (24,000 lbs) of recovered yellowcake more than compensates for the cost of this control system. The low energy venturi scrubber is 1.6% more efficient that the wet impingement scrubber and will recover an estimated additional 200 kilograms (440 lbs) of yellowcake per year. The value of this additional recovered yellowcake is approximately equal to the increased annual operating costs of the low energy venturi scrubber as compared to the wet impinger. The present worth of these systems are, therefore, not included as a control cost for the model mill.

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Table 9.0-1

Radiological Impact of Airborne Effluents versus Control Costs for a Model Uranium Mill

| Controls | Source Term[a] | Maximum Lung Dose to an Individual[b] | Present Worth |
|---|---|---|---|
| (Table 8.1) | (mCi/yr) | (mrem/yr) | (1974 $/facility) |
| None | >20,000 | >20,000 | 0 |
| A1; B1[c][d] | 205 | 200 | 172,000 |
| A1; B2[d] | 75 | 73 | 172,000 |
| A1; B3 | 35 | 34 | 262,000 |
| A2; B3 | 25 | 24 | 290,000 |
| A2; B3; C2 | 15 | 15 | 432,000 |
| A2; B4; C2 | 6 | 6 | 561,000 |
| A3; B4; C2 | 1.5 | 1.5 | 701,000 |
| A4; B4; C2 | 0.3 | 0.3 | 867,000 |
| C1[c] | 0 | 0 | 2,750,000 |

[a]Alpha emitting radionuclides as insoluble, respirable particulate matter.
[b]For the assumed worst case of an individual permanently occupying a location exhibiting a $\chi/Q$ of $6 \times 10^{-6}$ s/m$^3$.
[c]Assumed current level of controls for new mills.
[d]Costs for control options B1 and B2 not included, since they are more than compensated by the value of product recovered.

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## 10.0 Retrofitting Control Technology to Operating Uranium Mill

The cost and practicality of retrofitting control technology systems to an operating uranium mill was not included in Reference (6). The cost is judged to be approximately the same order of magnitude as the cost to install the same control system in a new mill.

The cost and practicality of retrofitting control measures to operational tailings piles that do not use clay core dam impoundment technologies must be considered on an individual basis.

# REFERENCES

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