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Nuclear Dynamics Consequence Analysis (NDCA) for the Disposal of Spent Nuclear Fuel in an Underground Geologic Repository

Volume 3: Appendices

Idaho National Engineering and Environmental Laboratory
Lockheed Martin Idaho Technologies Company
Idaho Falls, Idaho 83415

October 1998

This report was not produced under a quality assurance program that satisfies the requirements of the National Spent Nuclear Fuel Program and DOE/RW-0333P, Office of Civilian Radioactive Waste Management Quality Assurance Requirements Description. Therefore, the data in the report are not considered qualified and are not to be relied upon to address safety and waste isolation issues until an accepted qualification process has been completed.

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CONTENTS

APPENDICES

Figures

Tables

APPENDIX A
Acronyms and Abbreviations

BRAGFLO_T	-	Brine (Groundwater) and Gas Flow In Thermal in Hydrologic Code
CDF	-	Cumulative Distribution Function
CCDF	-	Complementary Cumulative Distribution Function
CFR	-	Code of Federal Regulations
CX	-	Static Criticality
DINO	-	Fully coupled Nuclear Dynamics and Thermal Hydrologic Code
DOE	-	United States of America Department of Energy
DOE-EM	-	Department of Energy / Office of Environmental Management
DOE-RW	-	Department of Energy / Office of Radioactive Waste
DSNF	-	Defense Spent Nuclear Fuel
DHLW	-	Defense High-Level Radioactive Waste
DTC	-	Doppler Temperature Coefficient
DTHX	-	Nuclear Dynamics coupled with Thermal Hydrology Model
DX	-	Nuclear Reactor Dynamics
ENDF	-	Evaluated Nuclear Data File
EPA	-	Environmental Protection Agency
FEPs	-	Features, Events, and Processes
FYP	-	Fission Yield Product
GTC	-	Ground Temperature Conditions
GW	-	Groundwater
HEU	-	Highly Enriched Uranium
HLW	-	High-Level radioactive Waste
INEL	-	Idaho National Engineering Laboratories (Acronym prior to INEEL)
INEEL	-	Idaho National Engineering and Environmental Laboratory
KX	-	Nuclear Kinetics
LANL	-	Los Alamos National Laboratories
LASL	-	Los Alamos Scientific Laboratory (Acronym prior to LANL)
LEU	-	Low-Enriched Uranium
LMFBR	-	Liquid Metal Fast Breeder Reactor
LMITCO	-	Lockheed Martin Idaho Technologies Companies, Inc.
MB	-	Megabytes
MCNP	-	Monte Carlo N-Particle (previously Monte Carlo code for Neutron and Photon transport)
MDS	-	Multi-Dimensional Sensitivity
M/F	-	Moderator to Fissile ratio
MFP	-	Mean Free Path (for a neutron)
MPC	-	Multi-Purpose Canister

MTHM	-	Metric Tons Heavy Metal (one metric ton = 1000 kg)
M&O	-	Maintenance and Operations
NARK	-	NucleAr Reactor Kinetics code
NAWG	-	Nuclear Analysis Working Group
NDCA	-	Nuclear Dynamics Consequence Analysis
NF	-	Nordheim-Fuchs Approximation
NJOY	-	Neutron Cross-Section Library generation code
NSNFP	-	National Spent Nuclear Fuel Program
NWMP	-	Nuclear Waste Management Program
NWPA	-	Nuclear Waste Policy Act
ODE	-	Ordinary Differential Equations
ORIGEN	-	Oak Ridge National Laboratory Radionuclide Decay code
PA	-	Performance Assessment
PDF	-	Probability Density Function
PRA	-	Probabilistic Risk Analysis
QA	-	Quality Assurance
RAD	-	Radiation Absorbed Dose
RNCS	-	Repository Nuclear Code System
RKeff	-	Repository K-effective (Pre-and Post- Processor Code for MCNP)
RNAG	-	Repository Nuclear Analysis Group (located at SNL)
ROM	-	Rough-Order-of-Magnitude
RW	-	Radioactive Waste
$S(\alpha,\beta)$	-	Thermal Scattering Kernel
SAPHIRE	-	Fault Tree Analysis Code (for PRA)
SLAM	-	Monte Carlo Simulation Code (for PRA)
SNF	-	Spent Nuclear Fuel
SNL	-	Sandia National Laboratories
TFM	-	Thermal Fissile Material
THX	-	Thermal-Hydrology Model
TRU	-	Transuranic
UDX	-	Uncoupled Nuclear Dynamics Model
UNM	-	University of New Mexico
WPG	-	Weapons-Grade Plutonium
WIPP	-	Waste Isolation Pilot Plant
YMP	-	Yucca Mountain Project
YMS	-	Yucca Mountain Site
YR	-	Year

APPENDIX B

Background on Performance Assessment (PA) Process

B.1 Overview

The process of assessing the performance of any technological system is termed "performance assessment" (PA). However, the term has taken on a specialized meaning among risk analysts and systems engineers since it was first defined in the Environmental Protection Agency (EPA) Standards, 40 CFR Part 191 (EPA, 1985). Performance assessment in this context is a structured methodology for determining the compliance of geologic nuclear waste repositories with established federal rules and standards. Formally, PA can be defined as, "a structured plan of investigation in which (1) features, events, and processes (FEPs) associated with a system are identified and ranked according to the degree that they affect system behavior, (2) the effects of significant FEPs on system performance are examined by means of mathematical models, taking into account all known uncertainties in model formulations and model parameters, and (3) the results of system performance, including the uncertainty of those results, are presented in probability distributions for system performance measures," (EPA 1985). A detailed, qualitative discussion of the PA process is provided in Rechard 1995a, Tierney 1995, and Helton 1993.

With respect to nuclear criticality, the most important part of the PA methodology is the "FEPs screening". The FEPs screening determines if each identified FEP plays an essential role in the performance of the system. The screening process proceeds by asking the following three questions for each FEP:

(1) Is the presence or occurrence of the FEP highly improbable?

(2) If the FEP is assumed to be present or may occur at some time during the period of performance of the system, would its presence or action cause only minor changes in normal system performance?

(3) Does the FEP fall into a regulatory category allowing it to be eliminated?

If the answer to either question is "yes," the FEP and its effects can be ignored in the PA model-building process. The ultimate goal of the NDCA project is to provide the basis for FEPs screening arguments for post-closure nuclear criticality of spent nuclear fuel in the Yucca Mountain Repository without requiring criticality safety enhancements for the waste packages.

Figure B-1 identifies the overall relationship between nuclear criticality, repository performance assessment (PA) and the decision analysis tools that determine the minimum required accuracy (data quality objectives, DQOs) for the waste characterization of SNFs regarding their acceptance into a geologic repository. Figure B-1 shows that

(1) Nuclear criticality is comprised of two components:
  a) Consequences of a critical assembly nuclear excursion (power producing neutron chain reactions)
    (1) Newly generated heat

(2) Radionuclide inventory (addition of new fission products to a post-closure repository.

b) Probability (and associated frequency of occurrence) of a critical assembly

(2) Nuclear criticality is integrated into the repository PA process in one of two ways:

(a) Nuclear criticality consequences and probabilities are used in the screening process for features, events, and processes (FEPs).

(b) Nuclear criticality consequences are directly integrated into the PA consequence modeling if those consequences significantly influence the performance of the repository.

(3) If the PA results identify health risk from the repository, the following factors are included into the system life cycle costs:

a) risks (doses due to releases to the environment),

b) costs,

c) schedule (transportation, conditioning of fuel, etc.), and

d) system constraints (e.g., prearranged agreements with state governments for a prioritized shipment of SNF or high-level waste, HLW).

4) Decision analysis tools are used to identify which set of activities (i.e., a unique path through the decision event trees) results in the optimum combination of risks, costs, schedule, and constraints. Unlike simple cost/benefit analysis, which may use linear programming techniques, the decision analysis tools applied in this process are stochastic. Thus, life cycle costs are presented as probabilistic distributions, and confidence levels associated with life cycle costs are identifiable. From these cost distributions, quantifiable parameters, such as mean costs, probability of failure (e.g., probability of 50% cost overruns), probability of different states of transportation scheduling (e.g., bottlenecks due to shipping/receiving), and so on, become readily apparent. Also, since the mathematics for stochastic analysis is similar to that used for repository PA, the PA regression analysis techniques can also be used to identify the relative contribution of individual activities to the total system life cycle costs and their associated uncertainties. The analysis of activity sets can be performed through two major approaches:

(a) Prescriptive System Prioritization Method, in which only preselected sets of activities are considered for analysis. Thus these sets are analyzed in a diagnostic mode only (see Boak 1996; 1997, Harris 1996 and Helton 1996).

(b) Generalized Stochastic Cost/Risk Tools, in which prospective activities (and associated cost/risk distributions) can be generated and investigated as part of the life cycle cost analysis. Thus, analysis can be performed to identify necessary cost reductions to existing activity sets in order to make them viable approaches.

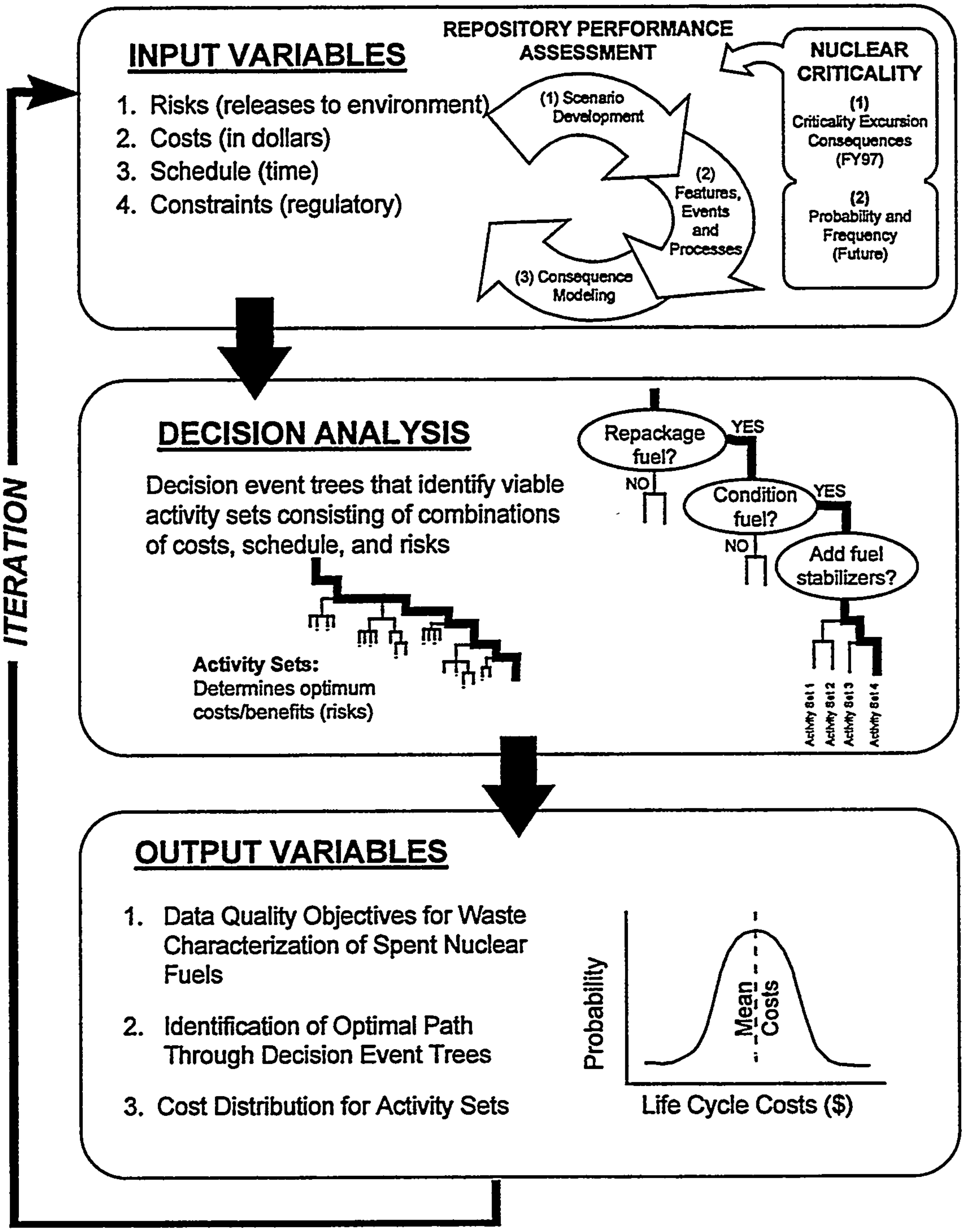


Figure B.1-1. Determination of optimal method for spent nuclear fuel disposal in a geological repository.

A detailed illustration of how nuclear criticality analysis is incorporated into the repository PA process is shown in Figure B.1-2. As can be seen, the preliminary steps in the process are the identification of the possible scenarios that could ultimately contribute to repository risks (i.e., releases from the repository to the accessible environment). There are various methods for scenario development (see Table B.1-1).

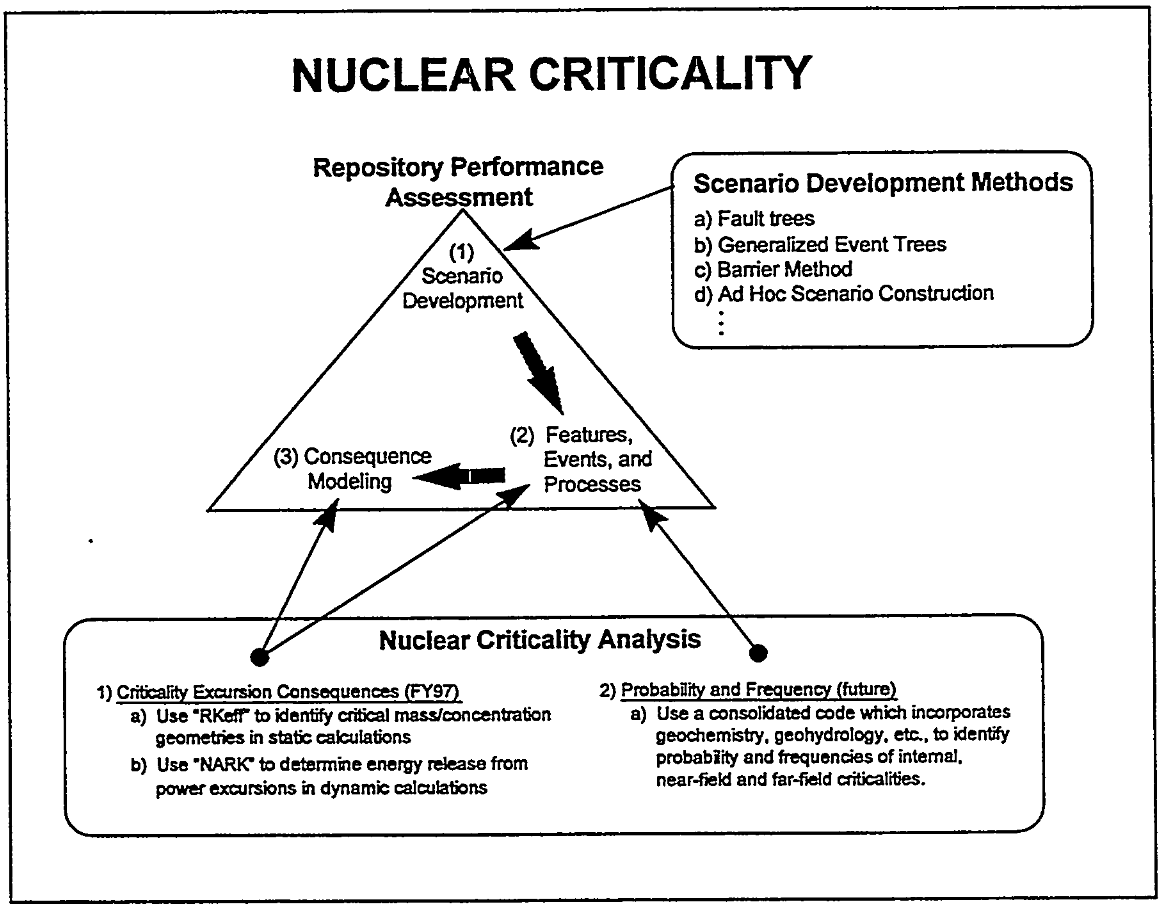


Figure B.1-2. How nuclear criticality analysis is incorporated into the PA process.

Table B.1-1. Scenario Development Methods

Method	Abbreviated Description
a) Fault Trees	Classical probabilistic fault trees can be used to identify indicators and probabilities of occurrence.
b) Generalized Event Trees	These trees are similar to fault trees, except that they only use "and" operations.
c) Barrier Event Trees	In this approach, the problem is reverse-engineered in that the barriers that must be overcome, for a repository to have significant releases, are identified.
d) Ad Hoc Scenario Construction	Often a special conditional scenario is developed. This "ad hoc" scenario identifies a specific set of events that may possibly (even if it is not probable) result in a significant release.

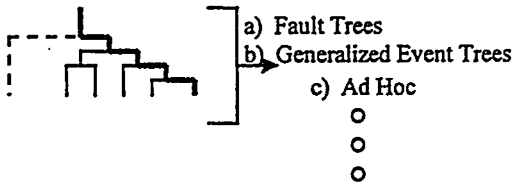
Figure B.1-2 also indicates the use of criticality excursion consequences and probability and frequency FEP screening arguments. As discussed in Section 1.1.3, FEPs can be screened out (i.e., identified as being reasonably insignificant) if the probability of occurrence is less than 10^{-4} over the regulatory time frame, the consequences are insignificant, or both. Obviously, if criticality has a significant cumulative effect on the releases from a repository, the consequences of criticality must become a standard subpart of the PA consequence modeling. The FEPs can also be screen based on their consequences. However, the regulations do not identify a limit for consequences that can readily be applied to criticality. Thus, if the consequences from nuclear excursions are not small enough to screen out the criticality FEPs or to satisfy the governing regulator, then the probability and frequency (yielding cumulative occurrences) will also be needed. This analysis could be difficult to perform if much detail is required. The most straightforward approach is to use PA computational results to identify initiating occurrences of nuclear excursions in the various repository regions: internal, near-field, and far-field. Follow-up calculations can then be used to determine the expected frequencies of re-occurrence of an excursion. This process would require use of many of the PA codes and significant computational resources. A possible approach that would streamline the analysis may be to model only the basic physics related to a criticality (i.e., corrosion mechanisms, groundwater transport, precipitation, etc.) in a single consolidated code. Thus, if the key input parameters are (Monte Carlo) sampled, the output database could be analyzed for the probabilities of internal, near-field, and far-field criticalities.

The last important feature of Figure B.1-2 is the inclusion of criticality consequences in the PA consequence modeling. Moverover, the PA can be computed by using various mathematical approaches as shown in Figures B.1-3 through B.1-6: (1) General Probabilistic Risk Approaches ("Simple" PA)[Figure B.1-3], (2) Parameter Estimation Performance Assessment ("Abstraction" PA)[Figure B.1-4], (3) Fully Stochastic Performance Assessment ("Complex" PA)[Figures B.1-2 and B.1-5], and (4) "Hybrid Abstraction" Approach (Figure B.1-6). Thus as illustrated in these figures, there are several approaches for PAs. The preferred approach would obviously be the simplest (level of treatment) that would identify compliance measures and is acceptable by the regulatory agency. Thus, the first attack on the problem may be a general probabilistic risk approach at any of the identified levels of treatment: identification of hazard, worst case, or quasi-worst case plausible upperbound, "best estimate" central value, probability and risk analysis, and display of risk uncertainties (Pate-Cornell, 1996).

A second approach would be the parameter estimation PA. The physics in this situation is decoupled, and the codes are run independently and parallel prior to Monte Carlo sampling. Codes are run in a sensitivity analysis and produce a results database; this is referred to as an abstraction. Later in the process, abstractions can be interpolated for the purpose of performing more calculations and producing more results. This approach can best be described as an "abstraction" of the PA.

A third approach, which is opposite of the parameter estimation PA, is Monte Carlo sampling. This method is more complex than the other two approaches, is a fully stochastic PA, and produces probabilistic results. Major codes run in a series after Monte Carlo sampling for all conditional scenarios. Flags are set for entire sets of codes so that each conditional scenario is separate. The physics is coupled throughout the scenarios, and each scenario produces its own computational results. These results are combined together to produce probabilistic results that are associated with confidence limits. A summary of PA mechanisms for consequence modeling is provided in Table B.1-2.

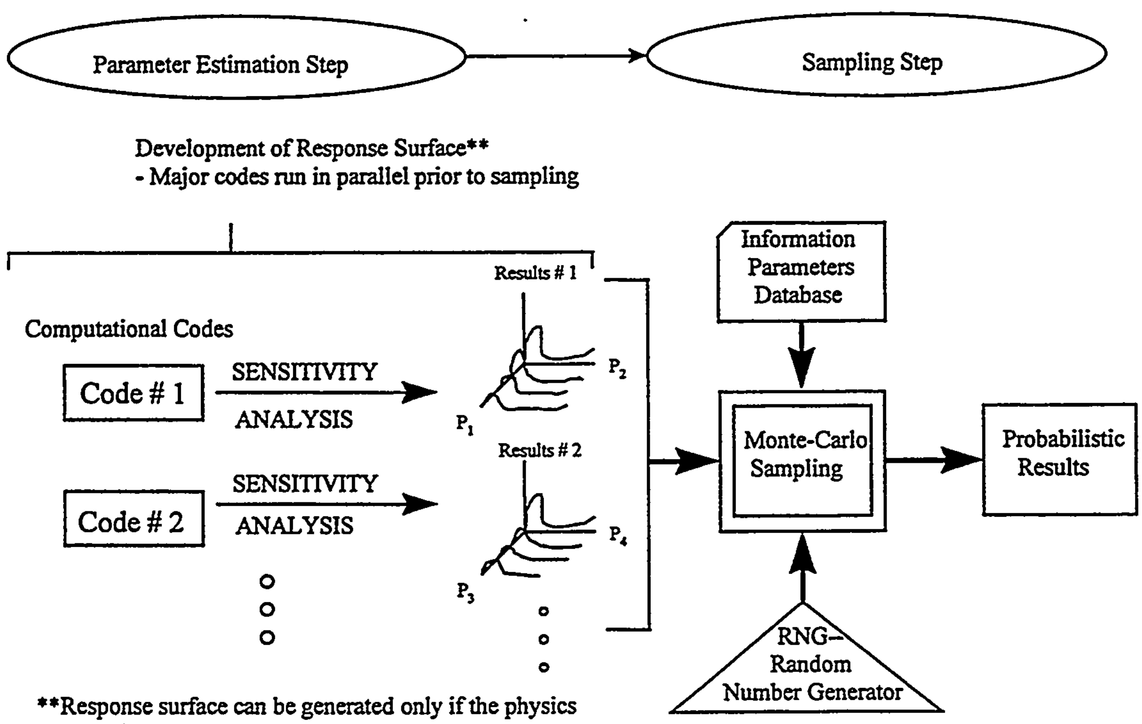
1.) General Probabilistic Risk Approaches



There are six levels in the treatment of uncertainty in risk analysis (see Ref. Pate-Cornell, 1996)*
1.) Identification of hazard / 4.) "Best estimate" central (?) value
2.) Worst case / 5.) Probability and risk analysis
3.) Quasi-worst case: Plausible upperbound / 6.) Display of risk uncertainties
* Many small-scale probabilistic risk analyses use the "Quasi-worst case" fault tree type approach.

Figure B.1-3. Simple performance assessment approaches.

2.) Parameter Estimation Performance Assessment

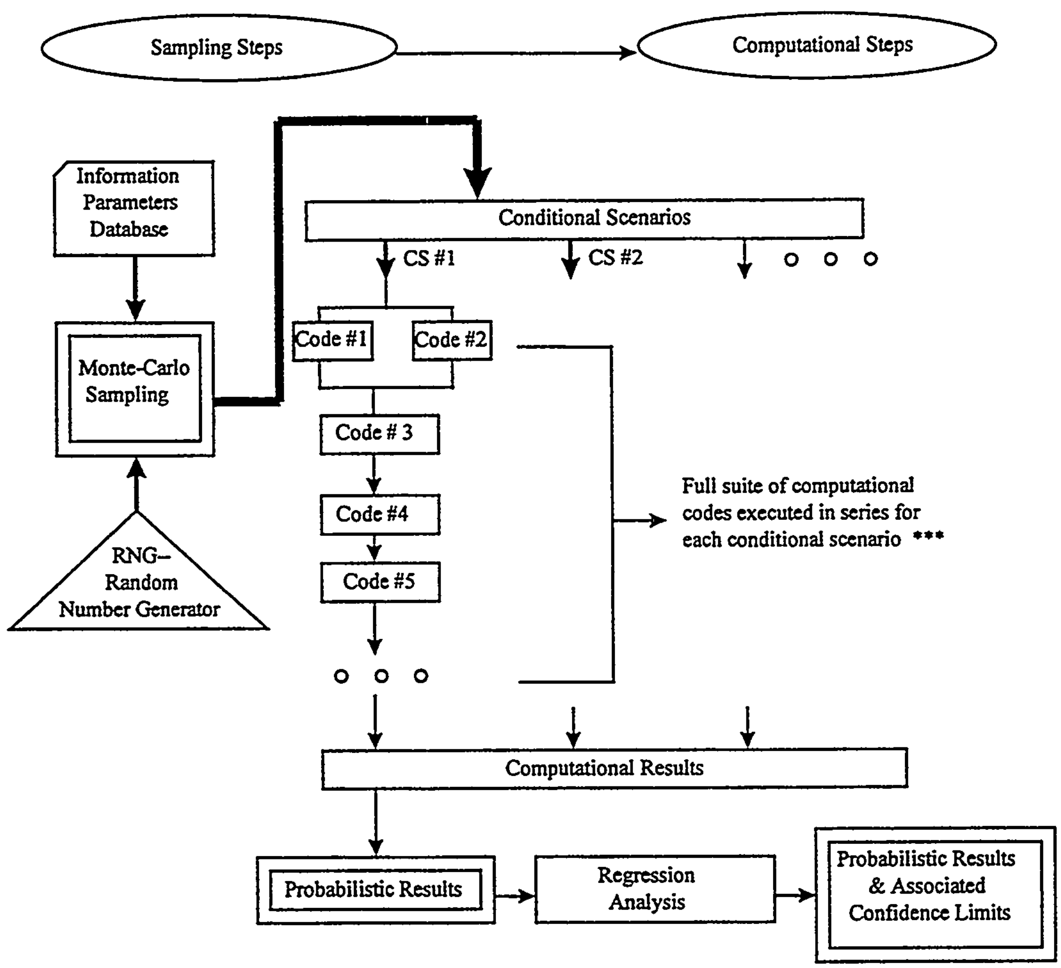


**Response surface can be generated only if the physics modeled by the codes are independent.

Figure B.1-4. Abstraction performance assessment approach.

3) Fully Stochastic Performance Assessment

-- Major codes run in series after sampling for all conditional scenarios

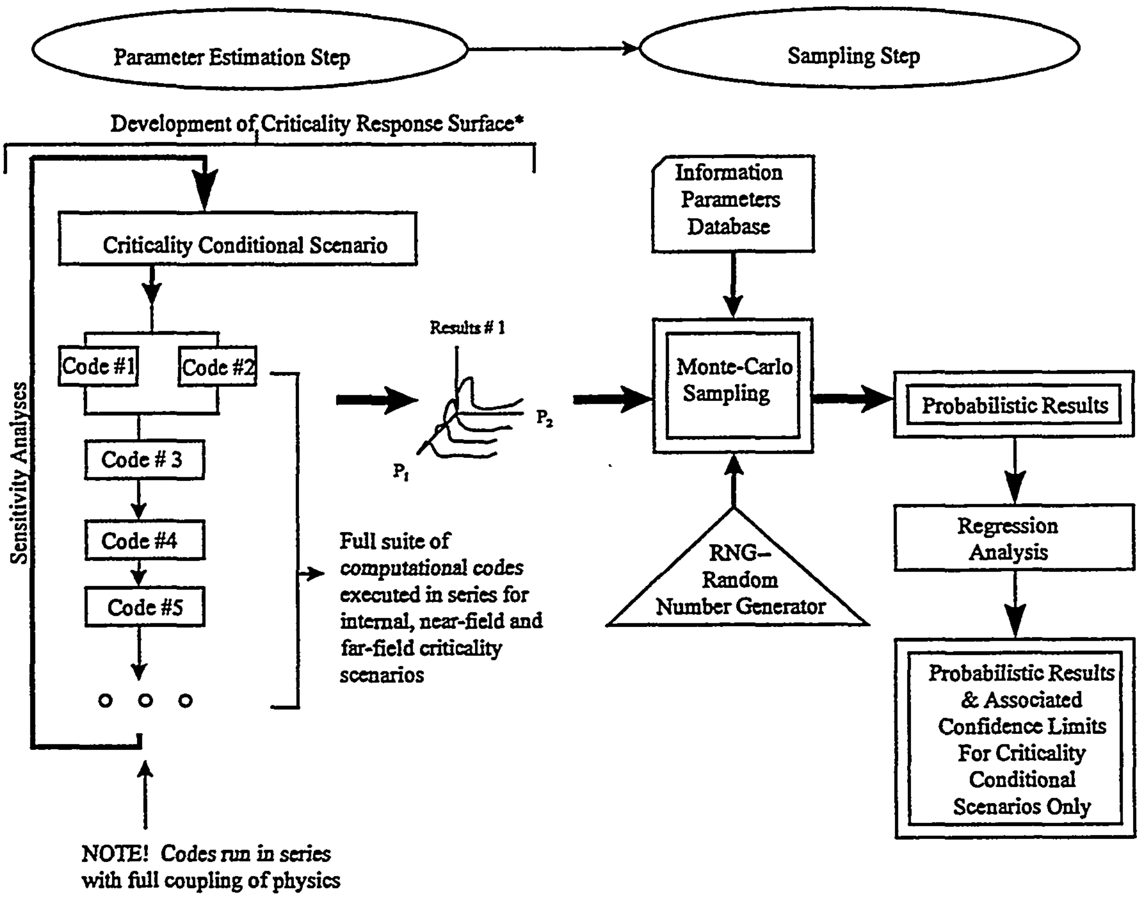


*** These codes must be executed in specific sequence to allow the proper coupling of the physic models (i.e., the codes/sub-models are not independent of each other).

Figure B.1-5. Complex performance assessment approach.

Hybrid Performance Assessment Mechanisms for Criticality Conditional Scenario, Modeling Using "Hybrid Abstraction"

-- i.e., using INEEL/PA codes (run in series) to generate an abstraction (response surface) for the conditional scenario of criticality



* These codes must be executed in specific sequence to allow the proper coupling of the physic models (i.e., the codes/sub-models are not independent of each other).

Figure B.1-6. Hybrid performance assessment mechanisms.

Table B.1-2 Performance Assessment Mechanisms for Consequence Modeling.

Major Category	Application			Result Types
	Simple Process[a]	Complex Preliminary Study	Process Detailed Study [b]	
1. Ad hoc, fault trees or generalized event trees	X	X		Conservative upper bound estimate of release
2. Stochastic PA assigning parameter estimation [c]		X	X	Probabilistic estimate of release
3. Stochastic PA using independent conditional scenarios [d]			X	Probabilistic estimates for release for all conditional scenarios and estimated uncertainties associated with calculated releases

a) Simple processes (SPs) may correspond to simple (small scale) analyses, such as the analysis of near-surface groundwater contamination because of leaking gasoline storage tanks at service stations.

b) Complex processes (CPs) may correspond to large-scale analysis, such as a geologic repository in which many conditional scenarios may exist for which the physics is complex (e.g., corrosion mechanics in a sealed repository would change the environment from aerobic to anaerobic).

c) An example of a PA using SP is the Yucca Mountain Project TSPA (Wilson, 1994) which uses existing computational results (in the form of response surfaces) in an interpolatory process to estimate the net results.

d) An example of a PA using CPs is Rechard 1998 which conducted detailed calculations for each sampled realization.

For a conditional scenario such as criticality, the hybrid performance can be used (see Figure B.1-6). This could be done by using the FY97 INEEL/PA codes as part of a small sensitivity analysis to generate an abstraction. The codes would be executed in a specific sequence to allow the proper coupling of the physics models for internal, near-field, and far-field scenarios. The output could be formulated as an abstraction. Once the sensitivity analysis has been completed and the results are found, the process is identical to parameter estimation PA.

B.2 Definition of Criticality in Performance Assessment

Previous discussion identifies that from a PA perspective, "criticality" is viewed in terms of "risk" (probability x consequences). The term "criticality" has different connotations to non-PA analysts and is defined differently in the literature, depending upon the specific discussion and intended context. Safety personnel define it as a subcritical limit for fissile mass accumulated in the system (and call it "criticality safety"). For these experts, the storage, handling, and transportation of fissile mass is the primary concern. In a repository, nuclear criticality safety constrains the design, maintenance, and operation of the repository more than other factors and disciplines involved in the repository analysis. As identified in 10 CFR 71, the transportation critical safety limit corresponds to the general limit of the neutron multiplication factor, k_{eff}, which is k_{eff} (and its associated bias and uncertainty error) ≤ 0.95. Even this simple safety requirement can be misleading to the general public since it is possible to identify fissile assemblies in which the fissile mass would need to be increased by at least an order of magnitude to produce $k_{eff} = 1.0$. In contrast, it is also possible to identify a situation wherein only a minor addition of fissile mass is needed to increase k_{eff} from 0.95 to 1.00. For application to the Yucca Mountain Repository, criticality requirements are identified in 10 CFR 60 and are discussed in Section 2.1.1. Unfortunately, the current requirements in 10 CFR 60 do not clearly specify whether the definition of "criticality" corresponds to "criticality safety" or "criticality consequences" (i.e., the results of a criticality excursion). This issue as been discussed in McLaughlin 1996 (Section 1.1.5).

The criticality consequences of a nuclear excursion in a repository may be comparable to those from previous criticality accidents involving small quantities of fissile material. These accidents resulted largely because of deficiencies in processes, equipment, maintenance, training, or supervision. Much of the responsibility for the series of six criticality accidents in the U.S. between 1958 and 1964, and the ten criticality accidents occurring between 1953 and 1960 and 1963 through 1978 in the former Soviet Union, were due to a lack of administrative control during a time of expanding production of enriched fissile material (see Table B.2-1). Criticality accidents occurring in aqueous systems in processing plants and, to a lesser degree, moderated metal and oxide systems are comparable to expected consequences of a nuclear excursion in a geologic repository. The processing plant accidents would involve critical systems that have significant moderation (near optimal moderation) because of water, and they would best resemble worst-case moderator situations produced by long-term corrosion processes resulting in the reconfiguration of fissile material. Of particular interest are the net consequences of the accident excursion, which indicate that the energy released (expressed in total and prompt fissions), while being deadly to humans in the immediate vicinity, is trivial with respect to additional fissions and mechanical damage capability (see Tables B.2-1 and B.2-2 for energy releases).

Table B.2-1. Criticality Accidents in Processing Plants.

USA*					
Date	**Plant**	**Total Fissions**	**Prompt Fissions**	**Doses (Rads)**	**Cause**
06/16/58	Y-12	1.3×10^{18}	7×10^{16}	365, 339, 327, 270, 236, 69, 69, and 23	U235 solution washed into drum
12/30/58	LASL	1.5×10^{17}	1.5×10^{17}	4400 (fatal), 135, and 3	Plutonium concentrated in solvent layer
10/16/59	ICPP	4×10^{19}	10^{17}	50 and 32 (primarily beta)	U235 solution siphoned into tank
1/25/61	ICPP	6×10^{17}	6×10^{17}	None	U235 solution forced into cylinder by air
04/7/62	Hanford Recuplex	8.2×10^{17}	10^{16}	87, 33, and 16	Plutonium solution in sump sucked into tank
07/24/64	Wood River Junction	1.3×10^{17}	10^{17}	10,000 (fatal), Two 60 to 100	U235 solution poured into tank
10/17/78	ICPP	3×10^{18}	Unknown	None	U235 buildup due to diluted scrub solution
UK*					
08/24/70	Windscale	10^{15}	10^{15}	Negligible	Plutonium concentrated in trapped solution
USSR**					
Date	**Plant**	**Total Fissions**	**Prompt Fissions**	**Doses (Rads)**	**Cause**
03/15/53	Mayak Enterprises, the Urals	2.5×10^{17}	N/A	100 and 1,000	Radioactive solution leaked from vessel during transfer within concrete cell
04/21/57	Mayak Enterprises, the Urals	2×10^{17}	N/A	One fatal, five debilitating	Uranium solution leaked from chamber
01/2/58	Mayak Enterprises, the Urals	2.3×10^{17}	N/A	Four fatal, one debilitating	HEUs ejected from holding tank
12/5/60	Mayak Enterprises, the Urals	10^{17}	N/A	5	Plutonium concentration too high in solution

Table B.2-1. Criticality Accidents in Processing Plants (Continued).

USSR**					
Date	**Plant**	**Total Fissions**	**Prompt Fissions**	**Doses (Rads)**	**Cause**
08/14/61	Siberian Chemical Combine	5×10^{15}, 10^{17}	N/A	200	Increase in temperature and equipment malfunction
09/7/62	Mayak Enterprises, the Urals	2×10^{17}	N/A	Insignificant	Plutonium left undissolved
01/30/63	Siberian Chemical Combine	7.9×10^{17}	N/A	Four 6 to 17	HEU solution was divided and transferred to different vessels
12/13/63	Siberian Chemical Combine	2.7×10^{17}	N/A	Insignificant	Vacuum valve to the trap was shut off
11/13/65	Electrostal', Fuel Fabrication Plant	10^{15}	N/A	3.5	Power accumulation in water reservoir
12/16/65	Mayak Enterprises, the Urals	7×10^{17}	N/A	≤ 0.03	Uranium mass exceeded safety Margin
12/10/68	Mayak Enterprises, the Urals	5×10^{17}	N/A	One fatal, one debilitating	Plutonium concentration too high
12/13/78	Siberian Chemical Combine	3×10^{18}	N/A	One 250 to 200, Seven 5 to 60	Plutonium mass in containers too high

* Taken from Paxton 1980. (Data values also found in Stratton, 1989).
** Taken from Frolov et al. 1995.

Table B.2-2. Criticality Accidents Involving Moderated Metal and Oxide Systems*.

Date	Plant	Total Fissions	Prompt Fissions	Doses (Rads)	Cause
06/06/45	LANL	4×10^{16}	3×10^{15}	66, 66, and 7.4	Water leaked into assembly
1950	Chalk River	Unknown	N/A	NA	Excess moderator added
06/02/52	LANL	1.22×10^{17}	N/A	135, 127, 60, and 9	Control removed, water not removed
12/12/52	Chalk River	1.2×10^{20}	N/A	Low	Positive void coefficient
07/22/54	INEEL	4.68×10^{18}	N/A	N/A	Planned transient extended
10/15/58	Vinca, Yugoslavia	2.6×10^{18}	N/A	205, 320, 410, 415, 422, and 433	Faulty power monitoring
03/15/60	Saclay, France	3×10^{18}	N/A	N/A	Removal of absorber rod
01/03/61	INEEL	4.4×10^{18}	N/A	3 fatalities	Removal of control rod
11/05/62	INEEL	1×10^{18}	N/A	N/A	Planned transient exceeded
12/30/65	Mol, Belgium	4.3×10^{17}	N/A	N/A	Inappropriate operation plus not draining tank
09/23/83	Buenos Aires, Argentina	4×10^{17}	N/A	Low	Failure to drain tank

*** Data from Stratton, 1989.**

When fission occurs in a fissile mass, whether it is caused by uranium or plutonium, the event results in the production of energy, fission fragments, neutrons, and various types of radiation (gamma, beta, etc.). For aboveground nuclear events, it is the prompt radiation production from fission that is of greatest concern because it could be lethal to humans in the immediate vicinity. In an underground repository, humans would not be present at the time when future corrosion/groundwater mechanisms could dilute, transport, and reconcentrate fissile material to possibly produce conditions yielding a nuclear excursion. Thus, the doses from post-closure prompt fission radiation would not be an issue of concern for geologic repository designs. The doses of concern would only be those delayed doses produced through the transport of the additional radioactive fission yield products to the accessible environment. The dose contribution due to additional excursions, as shown in the report, are insignificant in comparison to the dose contribution associated with the initial repository radionuclide inventory. Furthermore, by the time that sufficient fissile material has accumulate to cause a criticality, the water would probably have transported away significant quantities of the radionuclides from the original inventory.

The effects of prompt (mainly neutron and gamma) exposure on humans for a typical aqueous accident excursion (approximately 10^{17} prompt fissions) can result in a fatality for personnel within several meters of a critical assembly (fatalities are highly probable at distances less than three meters from an unshielded critical assembly, Knief 1985). The probability of a fatality drops off for large distances from the assembly (the spatial dependency of the exposure upon distance is due mainly to the geometrical spread of the radiation). Since there is significant shielding between post-closure fissile material in an underground geologic repository and the accessible environment, prompt exposures to humans are improbable and should not be an issue in the performance assessment of the repository. However, the regulatory compliance criteria for the proposed Yucca Mountain Repository remain uncertain; these criteria may be based entirely on release probabilities, risk to humans, or a combination of the two (Kastenberg, 1997).

APPENDIX C

Source Term

The initial source term for a consequence analysis of the proposed Yucca Mountain high-level radioactive waste repository (YMP) includes DOE-owned spent nuclear fuel (DSNF) and high-level waste forms (DHLWs), as well as commercially generated spent nuclear fuel (SNF). [1] The YMP is designed for a capacity of 75,000 metric tons of heavy metal (MTHM, 42 U. S. Code 10101 et seq), of which approximately 10%, or 7,000 MTHM, can be DSNF and DHLW. As this analysis shows, commercial SNF dominates the source term to an extent that minimizes the impact of DOE-owned waste.

As now envisioned, approximately a third of the DOE-owned waste, 2,831 MTHM, is DSNF and the balance, 6,960 MTHM, is DHLW. The DSNF include three categories of enrichment:

(1) low-enriched uranium (LEU): < 2% enrichment
(2) medium enriched uranium (MEU): 2% to 20 % enrichment
(3) highly enriched uranium (HEU): >20% enrichment

In the 1997 performance assessment (PA) of the defense wastes that are to be disposed in at YMP, the waste was categorized into sixteen types: 15 SNF categories and one HLW category (Rechard, 1997). The two commercial SNF categories are commercial pressurized water reactor and boiling water reactor fuels (see Table C.1-1).

Tables C.1-2 through C.1-11 presents various aspects of the projected YMP inventory: the MTHM, activity (curies) of the radionuclides in the inventory, fission yield products, fissile content, and some other parameters that may impact performance of the YMP repository. The radionuclide inventory, in all its aspects, is the fundamental component of a source term because the environmental standard for the repository defines the allowable releases to the environment. Both the radionuclide and the mass inventories are important to assessing performance. If the standard is dose-based, the allowed fraction of radionuclides released from the repository is determined in part by the allowed dose to a critical group of people or maximally exposed individual. If the primary vector for the material released from the repository is via groundwater transport, the material that can be transported from the repository by dissolution and related phenomena determines the subsurface release rate.

The following tables present inventories in units of radioactivity called "EPA units," which are obtained by normalizing the curie inventory of a particular radionuclide using 40 CFR Part 191, Appendix A, Table 1. Thus, the EPA units are risk measure.

[1] In this chapter, DSNF and DHLW are referred to collectively as "DOE-owned" SNF and HLW. Commercially generated SNF is referred to as "commercial SNF" and encompasses all commercially generated spent nuclear fuel.

A brief description of each table follows. The footnotes to the tables expand considerably on these brief definitions.

Table C.1-2 presents the MTHM content of the DOE-owned and commercial initial source term inventory. This table is based on the INEEL PA parameter database (Rechard, 1997). The table shows that the projected (radioactive) heavy metal inventory of YMP is significant: 65.79 x 10^3 MTHM, or 65.795 EPA units, although well within the 70,000 MTHM limit. However, the DSNF and DHLW comprise only 4.39% of the total MTHM inventory, considerably less than the 10% allowed.

Table C.1-3 compares the radionuclide inventory in curies for DOE-owned and initial commercial source term inventories projected for YMP. The total DOE-owned waste comprises only 1.45% of the total emplaced activity because of lower average fuel burnup; this proportion is reflected in the ten most abundant radionuclides. These ten most abundant radionuclides are the same for DOE-owned and commercial waste, although the order of abundance differs (e.g., ^{239}Pu is the sixth highest activity in DOE-owned waste but the eighth highest in commercial SNF).

Table C.1-4 provides a breakdown of the total projected YMP inventory, both commercial SNF and DOE-owned SNF and HLW, by radionuclide, and gives the inventory in both curies and EPA units. Note that the antepenultimate column of this table shows the release limits that are used to calculate EPA units from the curie inventories. As in Table C.1-3, the ten most abundant radionuclides comprise almost 70% of the inventory activity. Although ^{137}Cs and ^{90}Sr are the most abundant, comprising 42.3% of the activity, the plutonium isotopes and ^{241}Am are also abundant, providing 26.1% of the activity. It should be noted that ^{241}Pu, which is the second most abundant activity in Table C.1-3, is not among the "top ten" in terms of EPA units. EPA units are not calculated for ^{241}Pu because its half-life is less than 20 years, and it is thus not part of the source term EPA unit inventory (see Section 1.5 of this report and Appendix A of 40 CFR Part 191).

Table C.1-5 provides the same breakdown as Table C.1-3, only in EPA units instead of curies. EPA Units may be considered a dose surrogate (Rechard, 1995b). DOE-owned inventories comprises 1.14% of the total inventory in EPA units, as compared to 1.45% of the curies. The "top ten" radionuclides for the DOE-owned waste differ from those of the commercial SNF only in that ^{233}U is in the former list and ^{242m}Am is in the latter list.

Table C.1-6 presents the radionuclide inventory in terms of mass (kg), which is potentially important because solubility and corresponding concentrations depend on mass and not on radioactivity. The DOE-owned inventory is 4.39% of the total mass inventory (as compared to 1.45% of the curie inventory); this is still a relatively insignificant fraction of the total. The ten most abundant radionuclides are somewhat different for the DOE-owned versus the commercial inventory: ^{232}Th is third most abundant of the DOE radionuclides and not among the "top ten" of the commercial SNF, while ^{241}Am is one of the ten most abundant nuclides in commercial SNF but not in DOE-owned inventory. These are relatively very small differences, however, and should not affect repository performance. The two most abundant radionuclides by mass are, as expected, ^{238}U and ^{235}U. Since there is about 100 times as much ^{238}U as ^{235}U, ^{238}U can be expected to dominate the concentration of uranium in solution, with consequent dilution of dissolved radioactivity.

Table C.1-7 presents the same information as Table C.1-6, except that masses are in moles rather than mass, a useful presentation because solubility is usually presented as molarity or molality.

Table C-8 presents the uranium and plutonium fissile gram equivalents of the uranium isotopes and transuranic nuclides in the YMP inventory, as well as the fission yield products in the inventory (as a percentage of total fission product inventory).

Table C.1-9 presents the projected fissile inventory of the YMP expressed as ^{235}U fissile equivalents. The DOE-owned SNF and DHLW comprise 10.5 % of the ^{235}U-equivalent fissile mass inventory, as compared with 4.39% of the MTHM inventory, 1.45% of the curie inventory, and 1.14% of the EPA unit inventory. Although 10.5% is a significant fraction, it is still small enough that impact on repository performance will be small. The two largest contributors to fissile content are ^{239}Pu and ^{235}U, which together contribute about 99% of the total fissile inventory.

Table C.1-10 compares the uranium and plutonium-equivalent fissile enrichments for DOE-owned and commercial inventories. The uranium equivalent values are based on Table C.1-9, while the plutonium equivalent values are calculated using the ^{239}Pu fissile-gram-equivalent constants from Table C.1-8. Uranium/plutonium combinations are presented in ^{235}U fissile gram equivalents only. Although the total inventory appears to be 75.5% enriched in fissile plutonium isotopes, the overall average enrichment is 2.23% to 2.24%, indicating that plutonium is a relatively less important fissile component. The results in this table correlate with Table C.1-9: the 10.5% of the uranium-equivalent fissile inventory attributable to DOE-owned waste is reflected in the difference between the enrichment of the commercial SNF (e.g., 2.09%) and the YMP average enrichment (e.g., 2.23%). It is worth noting also that the plutonium fissile enrichment of commercial SNF is 75.4%

Table C.1-11 presents the fission load: total fissions back-calculated from the fission product inventory (assuming no decay time). The fission load is a measure of the number of fission events that the spent fuel has undergone and is reflected in the inventory of radionuclides that are fission products, and in their relative abundance not taking half-life into account. The fission load attributable to DOE-owned inventory is on the order of one percent of the total; 99% of the fissions are attributable to commercial SNF. The total inventory corresponds to approximately 6×10^{31} fissions.

The source term presented here provides a macroscopic view of the potential impacts of radionuclide and fissile inventory on repository performance and on criticality considerations. When the entire inventory is considered, the impact of the DSNF and DHLW appears to be almost insignificant compared to the impact of commercially generated SNF. This cannot be assumed for a single container of waste, a single SNF container, or even for a small, defined group of containers, however. Criticality and performance assessment considerations for any particular collection of high-level waste and/or SNF must be analyzed separately, using a distinct source term.

Table C.1-1. Spent Nuclear Fuel Categories from 1997 INEEL PA (taken from Rechard 1997).

No.	Waste Type	Represented by	Representative Cladding	Curies (2030 yr) (Total/ Category)	DSNF Total Mass of Heavy Metal (Total Mg/Category)a
1	Uranium metal	N-Reactor	Failed	1.35×10^7	2132
2	Uranium-zirconium alloy	Chicago Pile 5 (CP-5)	Zircaloy	5.05×10^5	0.040
3	Uranium-molybdenum alloy	Fermi Reactor	Zircaloy	1.35×10^6	420.9b
4	Uranium oxide-intact clad	Shippingport	Zircaloy	C4 $\times 10^7$	79.8
5	Uranium oxide-failed clad	Three Mile Island (TMI-2)	Zircaloy	2.91×10^7	88.3
6	Uranium aluminum alloy	Advanced Test Reactor (ATR)	Aluminum	5.01×10^7	9.42
7	Uranium silicate	Materials Testing Reactor (MTR)	Aluminum	1.27×10^7	12.7
8	Uranium-thorium carbide-intact clad	Ft. St. Vrain	Graphite	2.59×10^7	24.7
9	Uranium-thorium carbide-failed clad	Peach Bottom	Graphite	4.65×10^6	1.61
10	Uranium/plutonium carbide	Sodium Reactor Experiment (SRE)	Stainless steel	2.45×10^5	0.057
11	Mixed oxide fuel	Fast Flux Test Facility (FFTF)	Stainless steel	1.58×10^7	3.96
12	Uranium-thorium oxide	Shippingport Light Water Breeder Reactor (LWBR)	Zircaloy	C6 $\times 10^6$	55.5
13	Uranium/zirconium hydride	Training, Research, and Isotope production-General Atomic (TRIGA)	Stainless steel	2.96×10^6	2.22
14	Commercial fuel, PWR	21-PWR (Pressurized Water Reactor)	Zircaloy	8.37×10^9	41,530
15	Commercial fuel, BWR	44-BWR (Boiling Water Reactor)	Zircaloy	3.70×10^9	21,640

a This column shows the approximate mass (Mg) of heavy metal (uranium, plutonium, and thorium) based on the radioisotope's inventory that was used in the calculations. In most cases, these values provide a lower bound on the MTHM reported in Appendices A and B; however, errors are likely present in the ORIGEN2 inventories supplied by INEEL. Because the inventory determines the heat load and source term for the calculation, the 1997 PA's estimate of Mg was used in the calculations to be consistent with the radioisotope inventory. A comparison of reported and calculated values is presented in Section 3.1.17 of Rechard 1997.

b The most significant difference occurs between reported MTHM and mass of heavy metal in the inventory for Category 3. The calculated inventory of heavy metal is 420.9 Mg, while the reported amount (Appendices A and B of Rechard 1997) is 3.93 Mg.

Table C.1-2. YMP-Scale Metric Tons of Heavy Metals (MTHM) Inventory (Calendar Year = 2035 , Time = 0 yr) (a)

Category	YMP Metric Tons of Heavy Metals (MTHM) Inventory			
	Material Type	Packages	MTHM/Package	MTHM
DOE-Owned Spent Nuclear Fuels & Defense High-Level Wastes				
1	Uranium-metal/Zirconium	118	1.8065E+01	2.1317E+03
2	Uranium-Zirconium/Zirconium	9	4.2673E-03	3.8406E-02
3	Uranium-Molybdenum/Zirconium	55	7.6517E+00	4.2084E+02
4	Uranium-Oxide/Zirconium/ss	203	3.9133E-01	7.9440E+01
5	Uranium-Oxide/Al	595	1.4800E-01	8.8060E+01
6	Uranium-Alx/Al	750	1.2102E-02	9.0765E+00
7	Uranium-Silicon	255	4.9579E-02	1.2643E+01
8	Uranium/Thorium C(hi)/G	545	4.4779E-02	2.4405E+01
9	Uranium/Thorium C(li)/G	103	1.5615E-02	1.6083E+00
10	Uranium&Uranium;Pu C/non-G	5	1.1309E-02	5.6545E-02
11	MOx/Zirconium-ss	352	1.1158E-02	3.9276E+00
12	Uranium/Thorium oxide/Zr-ss	69	8.0391E-01	5.5470E+01
13	Uranium-Zirconium-Hx/ss,inc	102	2.1609E-02	2.2041E+00
16	High-Level Waste (b)	—	5.8203E-01	5.8203E+01
	Sub-Total	3,161	<0.88553> -ave- (c)	2.8877E+03 (4.39%) (d)
Commercial Spent Nuclear Fuels				
14	Commerical-PWR	4820	8.5784E+00	4.1348E+04
15	Commerical-BWR	2859	7.5410E+00	2.1560E+04
	Sub-Total	7,679	<8.1922> -ave- (e)	6.2908E+04 (95.61%) (f)
	Total	10,840		6.5795E+04
Units of Waste (g) = 6.5795E+04 / 1.0E+03 = 65.795				

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997).

(b) DHLW inventory is actually intermixed with DSNF in categories 2 → 13, "100 packages" is used for computational purposes only.

(c) Note, the average MTHM loading per package for DOE-owned inventory (DSNF & DHLW) is about an order of magnitude less than that for commercial SNF.

(d) Note, the maximum allowed MTHM of DOE-owned inventory is 7,000 MTHM (Ref. NWPA 1983) and DOE-owned inventory contribute to only 4.39% of the "Unit of Waste".

(e) Note, the average MTHM loading per package for commercial SNF is very large because current designs for the 54 inch diameter package allow up to 21 PWRs or 44 BWRs per package.

(f) Note, the maximum allowed MTHM of commercial Spent Nuclear Fuels is 70,000 MTHM (Ref. NWPA 1983) and commercial SNF contribute to 95.61% of the "Unit of Waste".

(g) Unit of Waste (as termed "Waste Unit Factor") calculated in accordance with Notes 1(a) and 1(b) of Table 1 of Appendix A of 40CFR191 (see Ref. EPA 1985).

Table C.1-3. YMP-Scale Radionuclide Inventory (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[Ci]	[%]	[Ci]	[%]	[Ci]	[%]	[Ci]	[%]	[Ci]
Ac227	3.07E+01	(8.92E+01)	2.83E+00	(8.23E+00)	6.07E-01	(1.76E+00)	2.70E-01	(7.85E-01)	3.44E+01
Ag108m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Am241	1.06E+06	(5.07E-01)	4.07E+05	(1.94E-01)	1.42E+08	(6.79E+01)	6.58E+07	(3.14E+01)	2.09E+08
Am242m	8.66E+02	(5.68E-02)	9.69E+01	(6.35E-03)	1.01E+06	(6.62E+01)	5.15E+05	(3.37E+01)	1.53E+06
Am243	1.81E+03	(1.08E-01)	1.73E+02	(1.03E-02)	1.17E+06	(6.96E+01)	5.09E+05	(3.03E+01)	1.68E+06
C14	9.16E+02	(9.76E-01)	0.00E+00	(0.00E+00)	5.93E+04	(6.32E+01)	3.36E+04	(3.58E+01)	9.38E+04
Cl36	4.36E+00	(5.86E-01)	0.00E+00	(0.00E+00)	4.96E+02	(6.66E+01)	2.44E+02	(3.28E+01)	7.44E+02
Cm243 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Cm244	7.72E+04	(6.75E-02)	5.36E+04	(4.69E-02)	8.19E+07	(7.16E+01)	3.23E+07	(2.83E+01)	1.14E+08
Cm245	3.04E+00	(1.34E-02)	2.65E-01	(1.16E-03)	1.68E+04	(7.38E+01)	5.95E+03	(2.62E+01)	2.28E+04
Cm246	5.15E+00	(1.09E-01)	3.01E-02	(6.38E-04)	3.55E+03	(7.53E+01)	1.16E+03	(2.46E+01)	4.72E+03
Cs135	2.04E+02	(5.74E-01)	5.41E+02	(1.52E+00)	2.39E+04	(6.72E+01)	1.09E+04	(3.07E+01)	3.55E+04
Cs137	3.43E+07	(7.11E-01)	6.21E+07	(1.29E+00)	3.34E+09	(6.92E+01)	1.39E+09	(2.88E+01)	4.83E+09
I129	1.64E+01	(7.02E-01)	8.94E-03	(3.83E-04)	1.63E+03	(6.98E+01)	6.89E+02	(2.95E+01)	2.34E+03
Mo93 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Nb93m ‡	5.70E+02	(5.37E-01)	2.58E+03	(2.43E+00)	6.89E+04	(6.50E+01)	3.40E+04	(3.21E+01)	1.06E+05
Nb94	1.05E+01	(1.87E-02)	1.42E-01	(2.53E-04)	5.40E+04	(9.63E+01)	2.08E+03	(3.71E+00)	5.61E+04
Ni59	4.58E+02	(2.87E-01)	1.27E+02	(7.95E-02)	1.20E+05	(7.51E+01)	3.92E+04	(2.45E+01)	1.60E+05
Ni63	4.01E+05	(1.75E+00)	0.00E+00	(0.00E+00)	1.72E+07	(7.50E+01)	5.32E+06	(2.32E+01)	2.29E+07
Np237	1.62E+02	(5.55E-01)	1.33E+02	(4.55E-01)	2.12E+04	(7.26E+01)	7.72E+03	(2.64E+01)	2.92E+04
Pa231	8.67E+01	(9.29E+01)	4.58E+00	(4.91E+00)	1.39E+00	(1.49E+00)	6.20E-01	(6.65E-01)	9.33E+01
Pb210	7.90E-03	(3.15E+01)	1.28E-04	(5.11E-01)	1.17E-02	(4.67E+01)	5.32E-03	(2.12E+01)	2.50E-02
Pd107	2.84E+01	(3.33E-01)	0.00E+00	(0.00E+00)	5.93E+03	(6.96E+01)	2.56E+03	(3.01E+01)	8.52E+03
Pu238	6.00E+05	(2.65E-01)	1.88E+06	(8.29E-01)	1.64E+08	(7.23E+01)	6.03E+07	(2.66E+01)	2.27E+08
Pu239	4.34E+05	(1.77E+00)	2.23E+04	(9.08E-02)	1.71E+07	(6.96E+01)	7.00E+06	(2.85E+01)	2.46E+07
Pu240	3.14E+05	(8.76E-01)	1.55E+04	(4.33E-02)	2.50E+07	(6.98E+01)	1.05E+07	(2.93E+01)	3.58E+07
Pu241	6.20E+06	(1.67E-01)	6.96E+05	(1.88E-02)	2.53E+09	(6.83E+01)	1.17E+09	(3.16E+01)	3.71E+09
Pu242	2.57E+02	(1.90E-01)	2.36E+01	(1.74E-02)	9.21E+04	(6.81E+01)	4.29E+04	(3.17E+01)	1.35E+05
Ra226	7.88E-03	(8.73E+00)	4.41E-04	(4.89E-01)	5.64E-02	(6.25E+01)	2.55E-02	(2.83E+01)	9.02E-02
Ra228 ‡	8.50E+00	(1.00E+02)	0.00E+00	(0.00E+00)	8.82E-06	(1.04E-04)	3.69E-06	(4.34E-05)	8.50E+00
Se79	2.37E+02	(7.77E-01)	4.32E+02	(1.42E+00)	2.11E+04	(6.91E+01)	8.75E+03	(2.87E+01)	3.05E+04
Sm151	5.07E+05	(1.93E+00)	0.00E+00	(0.00E+00)	1.81E+07	(6.88E+01)	7.72E+06	(2.93E+01)	2.63E+07
Sn121m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Sn126	3.15E+02	(5.46E-01)	0.00E+00	(0.00E+00)	4.05E+04	(7.02E+01)	1.69E+04	(2.93E+01)	5.77E+04
Sr90	2.90E+07	(8.80E-01)	4.28E+07	(1.30E+00)	2.29E+09	(6.95E+01)	9.35E+08	(2.84E+01)	3.30E+09
Tc99	7.39E+03	(7.62E-01)	1.55E+04	(1.60E+00)	6.65E+05	(6.86E+01)	2.82E+05	(2.91E+01)	9.70E+05
Th229	2.73E+01	(9.97E+01)	7.10E-02	(2.59E-01)	1.48E-02	(5.40E-02)	5.55E-03	(2.03E-02)	2.74E+01
Th230	1.46E+00	(8.12E+00)	5.83E-02	(3.24E-01)	1.14E+01	(6.34E+01)	5.06E+00	(2.81E+01)	1.80E+01
Th232 ‡	8.71E+00	(9.46E+01)	4.94E-01	(5.37E+00)	1.45E-05	(1.58E-04)	6.00E-06	(6.52E-05)	9.20E+00

Table C.1-3. YMP-Scale Radionuclide Inventory (Continued) (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[Ci]	[%]	[Ci]	[%]	[Ci]	[%]	[Ci]	[%]	[Ci]
U232 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
U233	1.20E+04	(9.99E+01)	2.75E+00	(2.29E-02)	2.57E+00	(2.14E-02)	9.26E-01	(7.71E-03)	1.20E+04
U234	1.67E+03	(1.92E+00)	2.35E+02	(2.70E-01)	5.93E+04	(6.82E+01)	2.58E+04	(2.97E+01)	8.70E+04
U235	2.96E+02	(2.06E+01)	3.73E-01	(2.59E-02)	7.86E+02	(5.46E+01)	3.57E+02	(2.48E+01)	1.44E+03
U236	7.43E+02	(3.87E+00)	2.05E+00	(1.07E-02)	1.31E+04	(6.82E+01)	5.37E+03	(2.79E+01)	1.92E+04
U238	8.73E+02	(4.06E+00)	1.78E+01	(8.29E-02)	1.35E+04	(6.28E+01)	7.09E+03	(3.30E+01)	2.15E+04
Zr93	0.00E+00	(0.00E+00)	3.30E+03	(2.01E+00)	1.08E+05	(6.58E+01)	5.29E+04	(3.22E+01)	1.64E+05
Total	7.29E+07	(5.84E-01)	1.08E+08	(8.64E-01)	8.63E+09	(6.91E+01)	3.69E+09	(2.95E+01)	1.25E+10
	1.81E+08 (1.45 %) (b)				1.23E+10 (98.55 %) (c)				
Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)									
Cs137	3.43E+07	(7.11E-01)	6.21E+07	(1.29E+00)	3.34E+09	(6.92E+01)	1.39E+09	(2.88E+01)	4.83E+09
Pu241	6.20E+06	(1.67E-01)	6.96E+05	(1.88E-02)	2.53E+09	(6.83E+01)	1.17E+09	(3.16E+01)	3.71E+09
Sr90	2.90E+07	(8.80E-01)	4.28E+07	(1.30E+00)	2.29E+09	(6.95E+01)	9.35E+08	(2.84E+01)	3.30E+09
Pu238	6.00E+05	(2.65E-01)	1.88E+06	(8.29E-01)	1.64E+08	(7.23E+01)	6.03E+07	(2.66E+01)	2.27E+08
Am241	1.06E+06	(5.07E-01)	4.07E+05	(1.94E-01)	1.42E+08	(6.79E+01)	6.58E+07	(3.14E+01)	2.09E+08
Cm244	7.72E+04	(6.75E-02)	5.36E+04	(4.69E-02)	8.19E+07	(7.16E+01)	3.23E+07	(2.83E+01)	1.14E+08
Pu240	3.14E+05	(8.76E-01)	1.55E+04	(4.33E-02)	2.50E+07	(6.98E+01)	1.05E+07	(2.93E+01)	3.58E+07
Sm151	5.07E+05	(1.93E+00)	0.00E+00	(0.00E+00)	1.81E+07	(6.88E+01)	7.72E+06	(2.93E+01)	2.63E+07
Pu239	4.34E+05	(1.77E+00)	2.23E+04	(9.08E-02)	1.71E+07	(6.96E+01)	7.00E+06	(2.85E+01)	2.46E+07
Ni63	4.01E+05	(1.75E+00)	0.00E+00	(0.00E+00)	1.72E+07	(7.50E+01)	5.32E+06	(2.32E+01)	2.29E+07

Table C.1-3. YMP-Scale Radionuclide Inventory (Continued) (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory				
	DOE-Owned		Commercial		Total
	SNF	DHLW	PWR	BWR	
	[Ci] [%]	[Ci] [%]	[Ci] [%]	[Ci] [%]	[Ci]
Top 10 Radionuclides (Ranked on Inventory for DOE-Owned Wastes Only)					
Cs137	3.43E+07 (7.11E-01)	6.21E+07 (1.29E+00)	--	--	9.64E+07
Sr90	2.90E+07 (8.80E-01)	4.28E+07 (1.30E+00)	--	--	7.18E+07
Pu241	6.20E+06 (1.67E-01)	6.96E+05 (1.88E-02)	--	--	6.90E+06
Pu238	6.00E+05 (2.65E-01)	1.88E+06 (8.29E-01)	--	--	2.48E+06
Am241	1.06E+06 (5.07E-01)	4.07E+05 (1.94E-01)	--	--	1.47E+06
Pu239	4.34E+05 (1.77E+00)	2.23E+04 (9.08E-02)	--	--	4.56E+05
Sm151	5.07E+05 (1.93E+00)	0.00E+00 (0.00E+00)	--	--	5.07E+05
Ni63	4.01E+05 (1.75E+00)	0.00E+00 (0.00E+00)	--	--	4.01E+05
Cm244	7.72E+04 (6.75E-02)	5.36E+04 (4.69E-02)	--	--	1.31E+05
Pu240	3.14E+05 (8.76E-01)	1.55E+04 (4.33E-02)	--	--	3.30E+04
Top 10 Radionuclides (Ranked on Inventory for Commercial SNFs Only)					
Cs137	--	--	3.34E+09 (6.92E+01)	1.39E+09 (2.88E+01)	4.73E+09
Pu241	--	--	2.53E+09 (6.83E+01)	1.17E+09 (3.16E+01)	3.70E+09
Sr90	--	--	2.29E+09 (6.95E+01)	9.35E+08 (2.84E+01)	3.23E+09
Pu238	--	--	1.64E+08 (7.23E+01)	6.03E+07 (2.66E+01)	2.24E+08
Am241	--	--	1.42E+08 (6.79E+01)	6.58E+07 (3.14E+01)	2.08E+08
Cm244	--	--	8.19E+07 (7.16E+01)	3.23E+07 (2.83E+01)	1.14E+08
Pu240	--	--	2.50E+07 (6.98E+01)	1.05E+07 (2.93E+01)	3.55E+07
Sm151	--	--	1.81E+07 (6.88E+01)	7.72E+06 (2.93E+01)	2.58E+07
Pu239	--	--	1.71E+07 (6.96E+01)	7.00E+06 (2.85E+01)	2.41E+07
Ni63	--	--	1.72E+07 (7.50E+01)	5.32E+06 (2.32E+01)	2.25E+07

† Data values for radionuclides were previously reported in 1993 TSPA (Ref. TSPA 1993).

‡ Data values for radionuclides were not previously reported in 1993 TSPA (Ref. TSPA 1993).

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997). (In total 41 radionuclides are inventoried in the INEEL/PA-DB).

(b) Note, the total DOE-owned curie load is only 181. MCi. Thus only 1.45 % of the total curie load in YMP is due to DOE-Owned wastes.

(c) Note, the total Commercial curie load is 12,300. MCi. Thus 98.55 % of the total curie load in YMP is due to Commercial wastes.

Table C.1-4.
40CFR191 Release Limits and Source Term EPA Units for YMP-Scale DOE-SNF & Commercial-SNF (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide			YMP Waste				
ID (b)	Decay Mode (c)	Half-Life (c)	Total Inventory [Curies] (d)		Release Limits Inventory [Curies] (e)		Source EPA Unit
			DOE	COMM	(Ci/UW)	(Ci)	
Ac227	α,β^-,γ	21.77 a	3.35E+01	8.77E-01	100.	6579.5	5.23E-03
Ag108m	$ITe^-,\gamma,\varepsilon,\beta^+$	39.8 s	PR	PR			
Am241	α,γ,SF	432.7 a	1.47E+06	2.08E+08	100.	6579.5	3.18E+04
Am242m	α,γ,ITe^-,SF	141. a	9.63E+02	1.52E+06	100.	6579.5	2.32E+02
Am243	α,γ,SF	7.37E+03 a	1.98E+03	1.68E+06	100.	6579.5	2.55E+02
C14	β^-	5730 a	9.16E+02	9.29E+04	100.	6579.5	1.43E+01
Cf249	α,γ,SF	351 a	NR	NR	100.	6579.5	0.00E+00
Cf251	α,γ	9.0E+02 a	NR	NR	100.	6579.5	0.00E+00
Cl36	β^-	3.01E5 a	4.36E+00	1.13E-02			
Cm243	$\alpha,\gamma,SF,\varepsilon$	29.1 a	PR	PR	100.	6579.5	0.00E+00
Cm244	α,γ,SF	18.1 a	1.31E+05	1.14E+08			
Cm245	α,γ,SF	8.5E+03 a	3.31E+00	2.28E+04	100.	6579.5	3.46E+00
Cm246	α,γ,SF	4.76E+03 a	5.18E+00	4.71E+03	100.	6579.5	7.17E-01
Cs135	β^-	2.3E+06 a	7.45E+02	3.48E+04	1000.	65795.	5.40E-01
Cs137	β^-,γ	30.17 a	9.64E+07 (f)	4.73E+09 (g)	1000.	65795.	7.34E+04
Gd152	α	1.1E+14 a	NR	NR	100.	6579.5	0.00E+00
I129	β^-,γ	1.57E+07 a	1.64E+01	2.32E+03	100.	6579.5	3.55E-01
Mo93	ε,γ	3.5E3 a	PR	PR			
Nb93m	ITe^-	16.1 a	3.15E+03	1.03E+05			
Nb94	β^-,γ	2000. a	1.06E+01	5.61E+04	1000.	65795.	8.53E-01
Nd144	α	2.1E+15 a	NR	NR	100.	6579.5	0.00E+00
Ni59	ε	7.6E+04 a	5.85E+02	1.59E+05	1000.	65795.	2.43E+00
Ni63	β^-	100. a	4.01E+05	2.25E+07	1000.	65795.	3.48E+02
Np237	α,γ	2.14E+06 a	2.95E+02	2.89E+04	100.	6579.5	4.44E+00
Pa231	α,γ	3.28E+04 a	9.13E+01	2.01E+00	100.	6579.5	1.42E-02
Pb210	α,β^-,γ	22.3 a	8.03E-03	1.70E-02	100.	6579.5	3.81E-07
Pd107	β^-	6.5E+06 a	2.84E+01	8.49E+03	1000.	65795.	1.29E-01
Pu238	α,γ,SF	87.7 a	2.48E+06	2.24E+08	100.	6579.5	3.45E+04
Pu239	α,γ,SF	2.410E+04 a	4.56E+05	2.41E+07	100.	6579.5	3.73E+03
Pu240	α,γ,SF	6.56E+03 a	3.30E+05	3.55E+07	100.	6579.5	5.45E+03
Pu241	α,β^-,γ	14.4 a	6.90E+06	3.70E+09			
Pu242	α,γ,SF	3.75E+05 a	2.81E+02	1.35E+05	100.	6579.5	2.06E+01
Ra226	α,γ	1.60E+03 a	8.32E-03	8.19E-02	100.	6579.5	1.37E-05
Ra228	β^-,γ	5.76 a	8.50E+00	1.25E-05			
Se79	β^-	6.5E+04 a	6.69E+02	2.98E+04	1000.	65795.	4.64E-01
Sm147	α	1.06E+11 a	NR	NR	100.	6579.5	0.00E+00
Sm148	α	7.E+15 a	NR	NR	100.	6579.5	0.00E+00
Sm151	β^-,γ	90 a	5.07E+05	2.58E+07	1000.	65795.	4.00E+02
Sn121m	β^-,γ,ITe^-	55 a	PR	PR	1000.	65795.	0.00E+00
Sn126	β^-,γ	1.0E+05 a	3.15E+02	5.74E+04	1000.	65795.	8.77E-01
Sr90	β^-	29.1 a	7.18E+07	3.22E+09	1000.	65795.	5.01E+04
Tc99	β^-,γ	2.13E+05 a	2.29E+04	9.47E+05	10000.	657950.	1.47E+00

Table C.1-4. (Continued)
40CFR191 Release Limits and Source Term EPA Units for YMP-Scale DOE-SNF & Commercial-SNF (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide			YMP Waste				
ID (b)	Decay Mode (c)	Half-Life (c)	Total Inventory [Curies] (d)		Release Limits Inventory [Curies] (e)		Source EPA Unit
			DOE	COMM	(Ci./UW)	(Ci)	
Th229	α,γ	7.3E+03 a	2.74E+01	2.04E-02	100.	6579.5	4.16E-03
Th230	α,γ	7.54E+04 a	1.52E+00	1.65E+01	10.	657.95	2.73E-02
Th232	α,γ	1.4E+10 a	9.20E+00	2.05E-05	10.	657.95	1.40E-02
U232	α,γ,SF	70 a	PR	PR	100.	6579.5	0.00E+00
U233	α,γ,SF	1.592E+05 a	1.20E+04	3.50E+00	100.	6579.5	1.82E+00
U234	α,γ,SF	2.46E+05 a	1.90E+03	8.51E+04	100.	6579.5	1.32E+01
U235	α,γ,SF	7.04E+08 a	2.96E+02	1.14E+03	100.	6579.5	2.19E-01
U236	α,γ,SF	2.342E+07 a	7.45E+02	1.85E+04	100.	6579.5	2.92E+00
U238	α,γ,SF	4.47E+09 a	8.91E+02	2.06E+04	100.	6579.5	3.26E+00
Zr93	β^-,γ	1.5E+06 a	3.30E+03	1.61E+05	1000.	65795.	2.52E+00
		Sum =	1.81E+08 (1.45 %) (h)	1.23E+10 (98.6 %) (i)			2.88E+05 (j)
			1.25E+10				

Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)

Nuclide ID	Total Radionuclide Inventory [Curies]			Source Term EPA Units			
	DOE	COMM	Total	DOE	COMM	Total	Cum (%)
Cs137	9.64E+07 (f)	4.73E+09 (g)	4.83E+09	1.47E+03	7.19R+04	7.24E+04	25.1
Sr90	7.18E+07	3.22E+09	3.29E+09	1.09E+03	4.90E+04	4.95E+04	42.3
Pu238	2.48E+06	2.24E+08	2.26E+08	3.77E+02	3.41E+04	3.42E+04	54.2
Am241	1.47E+06	2.08E+08	2.09E+08	2.23E+02	3.16E+04	3.17E+04	65.2
Pu240	3.30E+05	3.55E+07	3.58E+07	5.01E+01	5.40E+03	5.44E+03	67.1
Pu239	4.56E+05	2.41E+07	2.46E+07	6.94E+01	1.06E+03	3.73E+03	68.4
Sm151	5.07E+05	2.58E+07	2.63E+07	7.71E+00	3.92E+02	4.00E+02	68.5
Ni63	4.01E+05	2.25E+07	2.29E+07	6.09E+00	3.42E+02	3.48E+02	68.7
Am243	1.98E+03	1.68E+06	1.68E+06	3.01E-01	2.55E+02	2.55E+02	68.7
Am242m	9.63E+02	1.52E+06	1.52E+06	1.47E-01	2.32E+02	2.32E+02	68.8

PR Not Reported in INEEL/PA Parameters Database (see Ref. INEEL_DB 1997), but reported in 1993 TSPA (Ref. TSPA 1993).

(a) Radionuclide inventory information taken from INEEL/Performance Assessment Parameters Database (see Ref. INEEL_DB 1997) (in total, 41 radionuclides are inventoried in the INEEL/PA-DB). (Note, YMP 1993 TSPA uses 39 radionuclides (Sr-90 and Cs-137 are considered non-contributors to release doses and are handled in separate calculations that accounts for their heat input to the stored waste package.)

(b) Radionuclides in bold are those 41 incorporated into the INEEL/PA-DB for analysis of releases due to sub-surface transport (Ref. INEEL_DB 1997, additional radionuclides may be identified in Appendix A of Ref. INEEL 1997).

(c) Decay mode and half-life information taken from the Chart of the Nuclides, 14th Ed. Ref. GE-1. [It is better for technical calculations, to use half-lives that are extracted from the databases of ORIGEN2 [Ref. OR-1] because the ORIGEN2 data are of a later version than that of Ref. GE-1 (see Ref. INEEL/PA-DB for ORIGEN2 values).]

(d) Total inventory (curie) data taken from Ref. INEEL_DB 1997. Values correspond to a "YMP-Scale" design basis.

(e) Release limits are determined in accordance with 40CFR191 (Appendix A, Table 1) [Ref. EPA 1985]. Left column corresponds to specific release limits (cumulative releases to the accessible environment for 10,000 years after disposal per "unit of waste" identified in Note 1(e) of Table 1, Appendix A, 40CFR191). Right column corresponds to release limit obtained for 65.795 Units of Waste (see Table C.1-2 for calculation of the Unit of Waste) determined from data in the INEEL/PA-DB [Ref. INEEL_DB 1997]. (Note, the 65.795 value for the Units of Waste corresponds only to data at calendar year 2035 (i.e., closure date of the YMP facility)).

(f) Isotope with dominant curie load for DOE-owned wastes.

(g) Isotope with dominant curie load for Commercial wastes.

(h) Note, the total DOE-owned curie load is only 181. MCi (also, 1.45 % of the total curie load in YMP). The average DOE-Owned curie load is 1.81E+08/3261 = 5.55E+04 (Ci/package) = **0.886 (MTHM/package)** for a total of 2,888. MTHM. [See Table C.1-2 for MTHM inventory. Also note that the limit for SNL-Owned waste in YMP is 7,000. MTHM (see Ref. NWPA 1983).]

(i) Note, the total Commercial curie load is 12,300. MCi (also, 98.6 % of the total curie load in YMP). The average Commercial curie load is 1.23E+10/7679 = 1.60E+06 (Ci/package) = **8.192 (MTHM/package)** for a total of 65,795. MTHM. [See Table C.1-2 for MTHM inventory. Also note that the limit for Commercial waste in YMP is 70,000. MTHM (see Ref. NWPA 1983).]

(j) The significance of the Source Term EPA Unit inventory of 2.88E+05 is that during the regulatory time (10,000 years as identified in 40CFR191, Ref. EPA) only one part in 288,000. (or 3.47 PPM) of the initial source term at closure is allowed to be released over the entire regulatory timeframe.

Table C.1-5. YMP-Scale Source Term EPA Unit Inventory (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP EPA Units Inventory (for Unit of Waste = 65.795) (b)								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[EPA]	[%]	[EPA]	[%]	[EPA]	[%]	[EPA]	[%]	[EPA]
Ac227	4.67E-03	(8.92E+01)	4.30E-04	(8.23E+00)	9.23E-05	(1.76E+00)	4.10E-05	(7.85E-01)	5.23E-03
Ag108m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Am241	1.61E+02	(5.07E-01)	6.19E+01	(1.94E-01)	2.16E+04	(6.79E+01)	1.00E+04	(3.14E+01)	3.18E+04
Am242m	1.32E-01	(5.68E-02)	1.47E-02	(6.35E-03)	1.54E+02	(6.62E+01)	7.83E+01	(3.37E+01)	2.32E+02
Am243	2.75E-01	(1.08E-01)	2.63E-02	(1.03E-02)	1.78E+02	(6.96E+01)	7.74E+01	(3.03E+01)	2.55E+02
C14	1.39E-01	(9.76E-01)	0.00E+00	(0.00E+00)	9.01E+00	(6.32E+01)	5.11E+00	(3.58E+01)	1.43E+01
Cl36	6.63E-05	(5.86E-01)	0.00E+00	(0.00E+00)	7.54E-03	(6.66E+01)	3.71E-03	(3.28E+01)	1.13E-02
Cm243 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Cm244	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Cm245	4.62E-04	(1.34E-02)	4.03E-05	(1.16E-03)	2.55E+00	(7.38E+01)	9.04E-01	(2.62E+01)	3.46E+00
Cm246	7.83E-04	(1.09E-01)	4.57E-06	(6.38E-04)	5.40E-01	(7.53E+01)	1.76E-01	(2.46E+01)	7.17E-01
Cs135	3.10E-03	(5.74E-01)	8.22E-03	(1.52E+00)	3.63E-01	(6.72E+01)	1.66E-01	(3.07E+01)	5.40E-01
Cs137	5.21E+02	(7.11E-01)	9.44E+02	(1.29E+00)	5.08E+04	(6.92E+01)	2.11E+04	(2.88E+01)	7.34E+04
I129	2.49E-03	(7.02E-01)	1.36E-06	(3.83E-04)	2.48E-01	(6.98E+01)	1.05E-01	(2.95E+01)	3.55E-01
Mo93 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Nb93m ‡	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	8.53E-01
Nb94	1.60E-04	(1.87E-02)	2.16E-06	(2.53E-04)	8.21E-01	(9.63E+01)	3.16E-02	(3.71E+00)	2.43E+00
Ni59	6.96E-03	(2.87E-01)	1.93E-03	(7.95E-02)	1.82E+00	(7.51E+01)	5.96E-01	(2.45E+01)	2.43E+00
Ni63	6.09E+00	(1.75E+00)	0.00E+00	(0.00E+00)	2.61E+02	(7.50E+01)	8.09E+01	(2.32E+01)	3.48E+02
Np237	2.46E-02	(5.55E-01)	2.02E-02	(4.55E-01)	3.22E+00	(7.26E+01)	1.17E+00	(2.64E+01)	4.44E+00
Pa231	1.32E-02	(9.29E+01)	6.96E-04	(4.91E+00)	2.11E-04	(1.49E+00)	9.42E-05	(6.65E-01)	1.42E-02
Pb210	1.20E-07	(3.15E+01)	1.95E-09	(5.11E-01)	1.78E-07	(4.67E+01)	8.09E-08	(2.12E+01)	3.81E-07
Pd107	4.32E-04	(3.33E-01)	0.00E+00	(0.00E+00)	9.01E-02	(6.96E+01)	3.89E-02	(3.01E+01)	1.29E-01
Pu238	9.12E+01	(2.65E-01)	2.86E+02	(8.29E-01)	2.49E+04	(7.23E+01)	9.16E+03	(2.66E+01)	3.45E+04
Pu239	6.60E+01	(1.77E+00)	3.39E+00	(9.08E-02)	2.60E+03	(6.96E+01)	1.06E+03	(2.85E+01)	3.73E+03
Pu240	4.77E+01	(8.76E-01)	2.36E+00	(4.33E-02)	3.80E+03	(6.98E+01)	1.60E+03	(2.93E+01)	5.45E+03
Pu241	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Pu242	3.91E-02	(1.90E-01)	3.59E-03	(1.74E-02)	1.40E+01	(6.81E+01)	6.52E+00	(3.17E+01)	2.06E+01
Ra226	1.20E-06	(8.73E+00)	6.70E-08	(4.89E-01)	8.57E-06	(6.25E+01)	3.88E-06	(2.83E+01)	1.37E-05
Ra228 ‡	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Se79	3.60E-03	(7.77E-01)	6.57E-03	(1.42E+00)	3.21E-01	(6.91E+01)	1.33E-01	(2.87E+01)	4.64E-01
Sm151	7.71E+00	(1.93E+00)	0.00E+00	(0.00E+00)	2.75E+02	(6.88E+01)	1.17E+02	(2.93E+01)	4.00E+02
Sn121m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Sn126	4.79E-03	(5.46E-01)	0.00E+00	(0.00E+00)	6.16E-01	(7.02E+01)	2.57E-01	(2.93E+01)	8.77E-01
Sr90	4.41E+02	(8.80E-01)	6.51E+02	(1.30E+00)	3.48E+04	(6.95E+01)	1.42E+04	(2.84E+01)	5.01E+04
Tc99	1.12E-02	(7.62E-01)	2.36E-02	(1.60E+00)	1.01E+00	(6.86E+01)	4.29E-01	(2.91E+01)	1.47E+00
Th229	4.15E-03	(9.97E+01)	1.08E-05	(2.59E-01)	2.25E-06	(5.40E-02)	8.44E-07	(2.03E-02)	4.16E-03
Th230	2.22E-03	(8.12E+00)	8.86E-05	(3.24E-01)	1.73E-02	(6.34E+01)	7.69E-03	(2.81E+01)	2.73E-02
Th232 ‡	1.32E-02	(9.46E+01)	7.51E-04	(5.37E+00)	2.20E-08	(1.58E-04)	9.12E-09	(6.52E-05)	1.40E-02

Table C.1-5. YMP-Scale Source Term EPA Unit Inventory (Continued)
(Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP EPA Units Inventory (for Unit of Waste = 65.795) (b)				
	DOE-Owned		Commercial		Total
	SNF	DHLW	PWR	BWR	
	[EPA] [%]	[EPA] [%]	[EPA] [%]	[EPA] [%]	[EPA]
U232 †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
U233	1.82E+00 (9.99E+01)	4.18E-04 (2.29E-02)	3.91E-04 (2.14E-02)	1.41E-04 (7.71E-03)	1.82E+00
U234	2.54E-01 (1.92E+00)	3.57E-02 (2.70E-01)	9.01E+00 (6.82E+01)	3.92E+00 (2.97E+01)	1.32E+01
U235	4.50E-02 (2.06E+01)	5.67E-05 (2.59E-02)	1.19E-01 (5.46E+01)	5.43E-02 (2.48E+01)	2.19E-01
U236	1.13E-01 (3.87E+00)	3.12E-04 (1.07E-02)	1.99E+00 (6.82E+01)	8.16E-01 (2.79E+01)	2.92E+00
U238	1.33E-01 (4.06E+00)	2.71E-03 (8.29E-02)	2.05E+00 (6.28E+01)	1.08E+00 (3.30E+01)	3.26E+00
Zr93	2.39E-02 (9.47E-01)	5.02E-02 (1.99E+00)	1.64E+00 (6.52E+01)	8.04E-01 (3.19E+01)	2.52E+00
Total	1.34E+03 (4.67E-01) (c)	1.95E+03 (6.77E-01) (d)	1.39E+05 (4.84E+01)	1.45E+05 (5.04E+01)	2.88E+05
	3.29E+03 (1.14 %)		2.84E+05 (98.9 %)		

Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)

Nuclide ID	SNF [EPA] [%]	DHLW [EPA] [%]	PWR [EPA] [%]	BWR [EPA] [%]	Total [EPA]
Cs137	5.21E+02 (7.11E-01)	9.44E+02 (1.29E+00)	5.08E+04 (6.92E+01)	2.11E+04 (2.88E+01)	7.34E+04
Sr90	4.41E+02 (8.80E-01)	6.51E+02 (1.30E+00)	3.48E+04 (6.95E+01)	1.42E+04 (2.84E+01)	5.01E+04
Pu238	9.12E+01 (2.65E-01)	2.86E+02 (8.29E-01)	2.49E+04 (7.23E+01)	9.16E+03 (2.66E+01)	3.45E+04
Am241	1.61E+02 (5.07E-01)	6.19E+01 (1.94E-01)	2.16E+04 (6.79E+01)	1.00E+04 (3.14E+01)	3.18E+04
Pu240	4.77E+01 (8.76E-01)	2.36E+00 (4.33E-02)	3.80E+03 (6.98E+01)	1.60E+03 (2.93E+01)	5.45E+03
Pu239	6.60E+01 (1.77E+00)	3.39E+00 (9.08E-02)	2.60E+03 (6.96E+01)	1.06E+03 (2.85E+01)	3.73E+03
Sm151	7.71E+00 (1.93E+00)	0.00E+00 (0.00E+00)	2.75E+02 (6.88E+01)	1.17E+02 (2.93E+01)	4.00E+02
Ni63	6.09E+00 (1.75E+00)	0.00E+00 (0.00E+00)	2.61E+02 (7.50E+01)	8.09E+01 (2.32E+01)	3.48E+02
Am243	2.75E-01 (1.08E-01)	2.63E-02 (1.03E-02)	1.78E+02 (6.96E+01)	7.74E+01 (3.03E+01)	2.55E+02
Am242m	1.32E-01 (5.68E-02)	1.47E-02 (6.35E-03)	1.54E+02 (6.62E+01)	7.83E+01 (3.37E+01)	2.32E+02

Table C.1-5. YMP-Scale Source Term EPA Unit Inventory (Continued) (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory				Total
	DOE-Owned		Commercial		
	SNF	DHLW	PWR	BWR	
	[EPA] [%]	[EPA] [%]	[EPA] [%]	[EPA] [%]	[EPA]
Top 10 Radionuclides (Ranked on Inventory for DOE-Owned Wastes Only)					
Cs137	5.21E+02 (7.11E-01)	9.44E+02 (1.29E+00)	--	--	1.47E+03
Sr90	4.41E+02 (8.80E-01)	6.51E+02 (1.30E+00)	--	--	1.09E+03
Pu238	9.12E+01 (2.65E-01)	2.86E+02 (8.29E-01)	--	--	3.77E+02
Am241	1.61E+02 (5.07E-01)	6.19E+01 (1.94E-01)	--	--	2.23E+02
Pu239	6.60E+01 (1.77E+00)	3.39E+00 (9.08E-02)	--	--	6.94E+01
Pu240	4.77E+01 (8.76E-01)	2.36E+00 (4.33E-02)	--	--	5.01E+01
Sm151	7.71E+00 (1.93E+00)	0.00E+00 (0.00E+00)	--	--	7.71E+00
Ni63	6.09E+00 (1.75E+00)	0.00E+00 (0.00E+00)	--	--	6.09E+00
U233	1.82E+00 (9.99E+01)	4.18E-04 (2.29E-02)	--	--	1.82E+00
Am243	2.75E-01 (1.08E-01)	2.63E-02 (1.03E-02)	--	--	3.01E-01
Top 10 Radionuclides (Ranked on Inventory for Commercial SNFs Only)					
Cs137	--	--	5.08E+04 (6.92E+01)	2.11E+04 (2.88E+01)	7.19E+04
Sr90	--	--	3.48E+04 (6.95E+01)	1.42E+04 (2.84E+01)	4.90E+04
Pu238	--	--	2.49E+04 (7.23E+01)	9.16E+03 (2.66E+01)	3.42E+04
Am241	--	--	2.16E+04 (6.79E+01)	1.00E+04 (3.14E+01)	3.16E+04
Pu240	--	--	3.80E+03 (6.98E+01)	1.60E+03 (2.93E+01)	5.40E+03
Pu239	--	--	2.60E+03 (6.96E+01)	1.06E+03 (2.85E+01)	3.66E+03
Sm151	--	--	2.75E+02 (6.88E+01)	1.17E+02 (2.93E+01)	3.92E+02
Ni63	--	--	2.61E+02 (7.50E+01)	8.09E+01 (2.32E+01)	3.42E+02
Am243	--	--	1.78E+02 (6.96E+01)	7.74E+01 (3.03E+01)	2.55E+02
Am242m	--	--	1.54E+02 (6.62E+01)	7.83E+01 (3.37E+01)	2.32E+02

† Data values for radionuclides were previously reported in 1993 TSPA (Ref. TSPA 1993).

‡ Data values for radionuclides were not previously reported in 1993 TSPA (Ref. TSPA 1993).

(a) SNF and HLW inventory data taken from INEEL/PA Parameters Database (Ref. INEEL_DB 1997, updated values available in Appendix A of Ref. INEEL 1997). (In total 41 radionuclides are inventoried in the INEEL/PA-DB).

(b) Unit of Waste (also termed "Waste Unit Factor") calculated in accordance with Notes 1(a) and 1(b) of Table 1 of Appendix A of 40CFR191 (see Ref. EPA 1985).

(c) Note, the total DOE-owned EPA Unit load is only 3,290 units. Thus only 1.14 % of the total EPA Unit load in YMP is due to DOE-Owned inventory.

(d) Note, the total commercial EPA Unit load is 288,000 units. Thus 98.8 % of the total EPA Unit load in YMP is due to Commercial inventory.

Table C.1-6. YMP-Scale Source Term Mass Inventory (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Mass Inventory (b)								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[kg]	[%]	[kg]	[%]	[kg]	[%]	[kg]	[%]	[kg]
Ac227	4.24E-04	(8.92E+01)	3.91E-05	(8.23E+00)	8.39E-06	(1.76E+00)	3.73E-06	(7.85E-01)	4.76E-04
Ag108m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Am241	3.09E+02	(5.07E-01)	1.19E+02	(1.94E-01)	4.14E+04	(6.79E+01)	1.92E+04	(3.14E+01)	6.10E+04
Am242m	8.27E-02	(5.68E-02)	9.25E-03	(6.35E-03)	9.64E+01	(6.62E+01)	4.92E+01	(3.37E+01)	1.46E+02
Am243	9.08E+00	(1.08E-01)	8.68E-01	(1.03E-02)	5.87E+03	(6.96E+01)	2.55E+03	(3.03E+01)	8.43E+03
C14	2.06E-01	(9.76E-01)	0.00E+00	(0.00E+00)	1.33E+01	(6.32E+01)	7.54E+00	(3.58E+01)	2.11E+01
Cl36	1.32E-01	(5.86E-01)	0.00E+00	(0.00E+00)	1.50E+01	(6.66E+01)	7.39E+00	(3.28E+01)	2.25E+01
Cm243 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Cm244	9.54E-01	(6.75E-02)	6.63E-01	(4.69E-02)	1.01E+03	(7.16E+01)	3.99E+02	(2.83E+01)	1.41E+03
Cm245	1.77E-02	(1.34E-02)	1.54E-03	(1.16E-03)	9.79E+01	(7.38E+01)	3.47E+01	(2.62E+01)	1.33E+02
Cm246	1.68E-02	(1.09E-01)	9.80E-05	(6.38E-04)	1.16E+01	(7.53E+01)	3.78E+00	(2.46E+01)	1.54E+01
Cs135	1.77E+02	(5.74E-01)	4.70E+02	(1.52E+00)	2.07E+04	(6.72E+01)	9.46E+03	(3.07E+01)	3.08E+04
Cs137	3.94E+02	(7.11E-01)	7.13E+02	(1.29E+00)	3.84E+04	(6.92E+01)	1.60E+04	(2.88E+01)	5.54E+04
I129	9.28E+01	(7.02E-01)	5.06E-02	(3.83E-04)	9.23E+03	(6.98E+01)	3.90E+03	(2.95E+01)	1.32E+04
Mo93 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Nb93m ‡	1.17E-07	(5.37E-01)	5.28E-07	(2.43E+00)	1.41E-05	(6.50E+01)	6.96E-06	(3.21E+01)	2.17E-05
Nb94	5.52E-02	(1.87E-02)	7.46E-04	(2.53E-04)	2.84E+02	(9.63E+01)	1.09E+01	(3.71E+00)	2.95E+02
Ni59	6.04E+00	(2.87E-01)	1.67E+00	(7.95E-02)	1.58E+03	(7.51E+01)	5.17E+02	(2.45E+01)	2.11E+03
Ni63	6.49E+00	(1.75E+00)	0.00E+00	(0.00E+00)	2.79E+02	(7.50E+01)	8.62E+01	(2.32E+01)	3.71E+02
Np237	2.30E+02	(5.55E-01)	1.89E+02	(4.55E-01)	3.01E+04	(7.26E+01)	1.10E+04	(2.64E+01)	4.15E+04
Pa231	1.84E+00	(9.29E+01)	9.70E-02	(4.91E+00)	2.94E-02	(1.49E+00)	1.31E-02	(6.65E-01)	1.98E+00
Pb210	1.03E-07	(3.15E+01)	1.68E-09	(5.11E-01)	1.53E-07	(4.67E+01)	6.97E-08	(2.12E+01)	3.28E-07
Pd107	5.52E+01	(3.33E-01)	0.00E+00	(0.00E+00)	1.15E+04	(6.96E+01)	4.97E+03	(3.01E+01)	1.65E+04
Pu238	3.51E+01	(2.65E-01)	1.10E+02	(8.29E-01)	9.58E+03	(7.23E+01)	3.52E+03	(2.66E+01)	1.33E+04
Pu239	6.98E+03	(1.77E+00)	3.59E+02	(9.08E-02)	2.75E+05	(6.96E+01)	1.13E+05	(2.85E+01)	3.95E+05
Pu240	1.38E+03	(8.76E-01)	6.80E+01	(4.33E-02)	1.10E+05	(6.98E+01)	4.61E+04	(2.93E+01)	1.57E+05
Pu241	6.02E+01	(1.67E-01)	6.76E+00	(1.88E-02)	2.46E+04	(6.83E+01)	1.14E+04	(3.16E+01)	3.60E+04
Pu242	6.73E+01	(1.90E-01)	6.18E+00	(1.74E-02)	2.41E+04	(6.81E+01)	1.12E+04	(3.17E+01)	3.54E+04
Ra226	7.97E-06	(8.73E+00)	4.46E-07	(4.89E-01)	5.71E-05	(6.25E+01)	2.58E-05	(2.83E+01)	9.13E-05
Ra228 ‡	3.63E-05	(1.00E+02)	0.00E+00	(0.00E+00)	3.77E-11	(1.04E-04)	1.58E-11	(4.34E-05)	3.63E-05
Se79	3.40E+00	(7.77E-01)	6.19E+00	(1.42E+00)	3.03E+02	(6.91E+01)	1.25E+02	(2.87E+01)	4.38E+02
Sm151	1.93E+01	(1.93E+00)	0.00E+00	(0.00E+00)	6.88E+02	(6.88E+01)	2.93E+02	(2.93E+01)	1.00E+03
Sn121m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Sn126	1.11E+01	(5.46E-01)	0.00E+00	(0.00E+00)	1.43E+03	(7.02E+01)	5.95E+02	(2.93E+01)	2.03E+03
Sr90	2.12E+02	(8.80E-01)	3.13E+02	(1.30E+00)	1.68E+04	(6.95E+01)	6.85E+03	(2.84E+01)	2.41E+04
Tc99	4.36E+02	(7.62E-01)	9.13E+02	(1.60E+00)	3.92E+04	(6.86E+01)	1.66E+04	(2.91E+01)	5.72E+04
Th229	1.28E-01	(9.97E+01)	3.34E-04	(2.59E-01)	6.96E-05	(5.40E-02)	2.61E-05	(2.03E-02)	1.29E-01
Th230	7.23E-02	(8.12E+00)	2.89E-03	(3.24E-01)	5.65E-01	(6.34E+01)	2.51E-01	(2.81E+01)	8.91E-01
Th232 ‡	7.94E+04	(9.46E+01)	4.51E+03	(5.37E+00)	1.32E-01	(1.58E-04)	5.47E-02	(6.52E-05)	8.39E+04

Table C.1-6. YMP-Scale Source Term Mass Inventory (Continued)
(Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Mass Inventory (b)				
	DOE-Owned		Commercial		Total
	SNF [kg] [%]	DHLW [kg] [%]	PWR [kg] [%]	BWR [kg] [%]	[kg]
U232 †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
U233	1.24E+03 (9.99E+01)	2.84E-01 (2.29E-02)	2.66E-01 (2.14E-02)	9.57E-02 (7.71E-03)	1.24E+03
U234	2.67E+02 (1.92E+00)	3.76E+01 (2.70E-01)	9.49E+03 (6.82E+01)	4.13E+03 (2.97E+01)	1.39E+04
U235	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	6.66E+05
U236	1.15E+04 (3.87E+00)	3.17E+01 (1.07E-02)	2.03E+05 (6.82E+01)	8.30E+04 (2.79E+01)	2.97E+05
U238	2.60E+06 (4.06E+00)	5.30E+04 (8.29E-02)	4.02E+07 (6.28E+01)	2.11E+07 (3.30E+01)	6.39E+07
Zr93	6.24E+02 (9.47E-01)	1.31E+03 (1.99E+00)	4.29E+04 (6.52E+01)	2.10E+04 (3.19E+01)	6.59E+04
Total	2.84E+06 (4.30E+00) (b)	6.23E+04 (9.44E-02) (c)	4.14E+07 (6.28E+01)	2.16E+07 (3.28E+01)	6.60E+07
	2.90E+06 (4.39 %)		6.31E+07 (95.6 %)		
Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)					
U238	2.60E+06 (4.06E+00)	5.30E+04 (8.29E-02)	4.02E+07 (6.28E+01)	2.11E+07 (3.30E+01)	6.39E+07
U235	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	6.66E+05
Pu239	6.98E+03 (1.77E+00)	3.59E+02 (9.08E-02)	2.75E+05 (6.96E+01)	1.13E+05 (2.85E+01)	3.95E+05
U236	1.15E+04 (3.87E+00)	3.17E+01 (1.07E-02)	2.03E+05 (6.82E+01)	8.30E+04 (2.79E+01)	2.97E+05
Pu240	1.38E+03 (8.76E-01)	6.80E+01 (4.33E-02)	1.10E+05 (6.98E+01)	4.61E+04 (2.93E+01)	1.57E+05
Th232 ‡	7.94E+04 (9.46E+01)	4.51E+03 (5.37E+00)	1.32E-01 (1.58E-04)	5.47E-02 (6.52E-05)	8.39E+04
Zr93	6.24E+02 (9.47E-01)	1.31E+03 (1.99E+00)	4.29E+04 (6.52E+01)	2.10E+04 (3.19E+01)	6.59E+04
Am241	3.09E+02 (5.07E-01)	1.19E+02 (1.94E-01)	4.14E+04 (6.79E+01)	1.92E+04 (3.14E+01)	6.10E+04
Tc99	4.36E+02 (7.62E-01)	9.13E+02 (1.60E+00)	3.92E+04 (6.86E+01)	1.66E+04 (2.91E+01)	5.72E+04
Cs137	3.94E+02 (7.11E-01)	7.13E+02 (1.29E+00)	3.84E+04 (6.92E+01)	1.60E+04 (2.88E+01)	5.54E+04

Table C.1-6. YMP-Scale Source Term Mass Inventory (Continued)
(Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[kg]	[%]	[kg]	[%]	[kg]	[%]	[kg]	[%]	[kg]
Top 10 Radionuclides (Ranked on Inventory for DOE-Owned Wastes Only)									
U238	2.60E+06	(4.06E+00)	5.30E+04	(8.29E-02)	–	–	–	–	2.65E+06
U235	1.37E+05	(2.06E+01)	1.73E+02	(2.59E-02)	–	–	–	–	1.37E+05
Th232 ‡	7.94E+04	(9.46E+01)	4.51E+03	(5.37E+00)	–	–	–	–	8.39E+04
U236	1.15E+04	(3.87E+00)	3.17E+01	(1.07E-02)	–	–	–	–	1.15E+04
Pu239	6.98E+03	(1.77E+00)	3.59E+02	(9.08E-02)	–	–	–	–	7.34E+03
Zr93	6.24E+02	(9.47E-01)	1.31E+03	(1.99E+00)	–	–	–	–	1.93E+03
Pu240	1.38E+03	(8.76E-01)	6.80E+01	(4.33E-02)	–	–	–	–	1.45E+03
Tc99	4.36E+02	(7.62E-01)	9.13E+02	(1.60E+00)	–	–	–	–	1.35E+03
U233	1.24E+03	(9.99E+01)	2.84E-01	(2.29E-02)	–	–	–	–	1.24E+03
Cs137	3.94E+02	(7.11E-01)	7.13E+02	(1.29E+00)	–	–	–	–	1.11E+03
Top 10 Radionuclides (Ranked on Inventory for Commercial SNFs Only)									
U238	–	–	–	–	4.02E+07	(6.28E+01)	2.11E+07	(3.30E+01)	6.13E+07
U235	–	–	–	–	3.64E+05	(5.46E+01)	1.65E+05	(2.48E+01)	5.29E+05
Pu239	–	–	–	–	2.75E+05	(6.96E+01)	1.13E+05	(2.85E+01)	3.88E+05
U236	–	–	–	–	2.03E+05	(6.82E+01)	8.30E+04	(2.79E+01)	2.86E+05
Pu240	–	–	–	–	1.10E+05	(6.98E+01)	4.61E+04	(2.93E+01)	1.56E+05
Zr93	–	–	–	–	4.29E+04	(6.52E+01)	2.10E+04	(3.19E+01)	6.39E+04
Am241	–	–	–	–	4.14E+04	(6.79E+01)	1.92E+04	(3.14E+01)	6.06E+04
Tc99	–	–	–	–	3.92E+04	(6.86E+01)	1.66E+04	(2.91E+01)	5.58E+04
Cs137	–	–	–	–	3.84E+04	(6.92E+01)	1.60E+04	(2.88E+01)	5.44E+04
Pu241	–	–	–	–	2.46E+04	(6.83E+01)	1.14E+04	(3.16E+01)	3.60E+04

† Data values for radionuclides were previously reported in 1993 TSPA (Ref. TSPA 1993).

‡ Data values for radionuclides were not previously reported in 1993 TSPA (Ref. TSPA 1993).

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997). (In total 41 radionuclides are inventoried in the INEEL/PA-DB).

(b) Mass inventory values calculated using half-lives from the Decay Libraries from ORIGEN2 (Ref. Croff 1980).

(c) Note, the total DOE-owned mass load (due to radionuclides) is only 2.90E+06 kg. **Thus only 4.39 % of the total mass load (due to radionuclides) in YMP is due to DOE-Owned inventory.**

(d) Note, the total commercial mass load (due to radionuclides) is 6.30E+07 kg. **Thus 95.6 % of the total mass load (due to radionuclides) in YMP is due to Commercial inventory.**

Table C.1-7. YMP-Scale Source Term Mole Inventory (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Mole Inventory (b)								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[mole]	[%]	[mole]	[%]	[mole]	[%]	[mole]	[%]	[mole]
Ac227	1.87E-03	(8.92E+01)	1.72E-04	(8.23E+00)	3.70E-05	(1.76E+00)	1.64E-05	(7.85E-01)	2.10E-03
Ag108m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Am241	1.28E+03	(5.07E-01)	4.92E+02	(1.94E-01)	1.72E+05	(6.79E+01)	7.96E+04	(3.14E+01)	2.53E+05
Am242m	3.42E-01	(5.68E-02)	3.82E-02	(6.35E-03)	3.98E+02	(6.62E+01)	2.03E+02	(3.37E+01)	6.02E+02
Am243	3.74E+01	(1.08E-01)	3.57E+00	(1.03E-02)	2.42E+04	(6.96E+01)	1.05E+04	(3.03E+01)	3.47E+04
C14	1.47E+01	(9.76E-01)	0.00E+00	(0.00E+00)	9.50E+02	(6.32E+01)	5.38E+02	(3.58E+01)	1.50E+03
Cl36	3.67E+00	(5.86E-01)	0.00E+00	(0.00E+00)	4.18E+02	(6.66E+01)	2.05E+02	(3.28E+01)	6.27E+02
Cm243 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Cm244	3.91E+00	(6.75E-02)	2.72E+00	(4.69E-02)	4.15E+03	(7.16E+01)	1.64E+03	(2.83E+01)	5.79E+03
Cm245	7.23E-02	(1.34E-02)	6.30E-03	(1.16E-03)	3.99E+02	(7.38E+01)	1.41E+02	(2.62E+01)	5.41E+02
Cm246	6.82E-02	(1.09E-01)	3.98E-04	(6.38E-04)	4.70E+01	(7.53E+01)	1.54E+01	(2.46E+01)	6.24E+01
Cs135	1.31E+03	(5.74E-01)	3.48E+03	(1.52E+00)	1.54E+05	(6.72E+01)	7.01E+04	(3.07E+01)	2.29E+05
Cs137	2.88E+03	(7.11E-01)	5.21E+03	(1.29E+00)	2.80E+05	(6.92E+01)	1.17E+05	(2.88E+01)	4.05E+05
I129	7.20E+02	(7.02E-01)	3.93E-01	(3.83E-04)	7.16E+04	(6.98E+01)	3.03E+04	(2.95E+01)	1.03E+05
Mo93 †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Nb93m ‡	1.26E-06	(5.37E-01)	5.68E-06	(2.43E+00)	1.52E-04	(6.50E+01)	7.49E-05	(3.21E+01)	2.34E-04
Nb94	5.87E-01	(1.87E-02)	7.94E-03	(2.53E-04)	3.02E+03	(9.63E+01)	1.16E+02	(3.71E+00)	3.14E+03
Ni59	1.02E+02	(2.87E-01)	2.84E+01	(7.95E-02)	2.69E+04	(7.51E+01)	8.77E+03	(2.45E+01)	3.58E+04
Ni63	1.03E+02	(1.75E+00)	0.00E+00	(0.00E+00)	4.43E+03	(7.50E+01)	1.37E+03	(2.32E+01)	5.90E+03
Np237	9.70E+02	(5.55E-01)	7.96E+02	(4.55E-01)	1.27E+05	(7.26E+01)	4.62E+04	(2.64E+01)	1.75E+05
Pa231	7.95E+00	(9.29E+01)	4.20E-01	(4.91E+00)	1.27E-01	(1.49E+00)	5.68E-02	(6.65E-01)	8.55E+00
Pb210	4.93E-07	(3.15E+01)	7.98E-09	(5.11E-01)	7.30E-07	(4.67E+01)	3.32E-07	(2.12E+01)	1.56E-06
Pd107	5.16E+02	(3.33E-01)	0.00E+00	(0.00E+00)	1.08E+05	(6.96E+01)	4.65E+04	(3.01E+01)	1.55E+05
Pu238	1.47E+02	(2.65E-01)	4.61E+02	(8.29E-01)	4.03E+04	(7.23E+01)	1.48E+04	(2.66E+01)	5.57E+04
Pu239	2.92E+04	(1.77E+00)	1.50E+03	(9.08E-02)	1.15E+06	(6.96E+01)	4.71E+05	(2.85E+01)	1.65E+06
Pu240	5.74E+03	(8.76E-01)	2.83E+02	(4.33E-02)	4.57E+05	(6.98E+01)	1.92E+05	(2.93E+01)	6.55E+05
Pu241	2.50E+02	(1.67E-01)	2.80E+01	(1.88E-02)	1.02E+05	(6.83E+01)	4.71E+04	(3.16E+01)	1.49E+05
Pu242	2.78E+02	(1.90E-01)	2.55E+01	(1.74E-02)	9.97E+04	(6.81E+01)	4.64E+04	(3.17E+01)	1.46E+05
Ra226	3.53E-05	(8.73E+00)	1.97E-06	(4.89E-01)	2.52E-04	(6.25E+01)	1.14E-04	(2.83E+01)	4.04E-04
Ra228 ‡	1.59E-04	(1.00E+02)	0.00E+00	(0.00E+00)	1.65E-10	(1.04E-04)	6.92E-11	(4.34E-05)	1.59E-04
Se79	4.31E+01	(7.77E-01)	7.85E+01	(1.42E+00)	3.83E+03	(6.91E+01)	1.59E+03	(2.87E+01)	5.55E+03
Sm151	1.28E+02	(1.93E+00)	0.00E+00	(0.00E+00)	4.56E+03	(6.88E+01)	1.94E+03	(2.93E+01)	6.63E+03
Sn121m †	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00	(0.00E+00)	0.00E+00
Sn126	8.81E+01	(5.46E-01)	0.00E+00	(0.00E+00)	1.13E+04	(7.02E+01)	4.73E+03	(2.93E+01)	1.61E+04
Sr90	2.36E+03	(8.80E-01)	3.49E+03	(1.30E+00)	1.87E+05	(6.95E+01)	7.62E+04	(2.84E+01)	2.69E+05
Tc99	4.40E+03	(7.62E-01)	9.24E+03	(1.60E+00)	3.96E+05	(6.86E+01)	1.68E+05	(2.91E+01)	5.78E+05
Th229	5.60E-01	(9.97E+01)	1.46E-03	(2.59E-01)	3.04E-04	(5.40E-02)	1.14E-04	(2.03E-02)	5.62E-01
Th230	3.14E-01	(8.12E+00)	1.26E-02	(3.24E-01)	2.46E+00	(6.34E+01)	1.09E+00	(2.81E+01)	3.87E+00
Th232 ‡	3.42E+05	(9.46E+01)	1.94E+04	(5.37E+00)	5.70E-01	(1.58E-04)	2.36E-01	(6.52E-05)	3.62E+05

Table C.1-7. YMP-Scale Source Term Mole Inventory (Continued)
(Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Mole Inventory				Total
	DOE-Owned		Commercial		
	SNF	DHLW	PWR	BWR	
	[mole] [%]	[mole] [%]	[mole] [%]	[mole] [%]	[mole]
U232 †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
U233	5.32E+03 (9.99E+01)	1.22E+00 (2.29E-02)	1.14E+00 (2.14E-02)	4.11E-01 (7.71E-03)	5.32E+03
U234	1.14E+03 (1.92E+00)	1.61E+02 (2.70E-01)	4.06E+04 (6.82E+01)	1.76E+04 (2.97E+01)	5.95E+04
U235	5.83E+05 (2.06E+01)	7.34E+02 (2.59E-02)	1.55E+06 (5.46E+01)	7.03E+05 (2.48E+01)	2.83E+06
U236	4.87E+04 (3.87E+00)	1.34E+02 (1.07E-02)	8.58E+05 (6.82E+01)	3.52E+05 (2.79E+01)	1.26E+06
U238	1.09E+07 (4.06E+00)	2.22E+05 (8.29E-02)	1.69E+08 (6.28E+01)	8.86E+07 (3.30E+01)	2.68E+08
Zr93	6.72E+03 (9.47E-01)	1.41E+04 (1.99E+00)	4.62E+05 (6.52E+01)	2.26E+05 (3.19E+01)	7.09E+05
Total	1.19E+07 (4.29E+00) (b)	2.82E+05 (1.01E-01) (c)	1.75E+08 (6.28E+01)	9.13E+07 (3.28E+01)	2.79E+08
	1.22E+07 (4.39 %)		2.66E+08 (95.6 %)		

Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)

Nuclide ID	SNF [mole] [%]	DHLW [mole] [%]	PWR [mole] [%]	BWR [mole] [%]	Total [mole]
U238	1.09E+07 (4.06E+00)	2.22E+05 (8.29E-02)	1.69E+08 (6.28E+01)	8.86E+07 (3.30E+01)	2.68E+08
U235	5.83E+05 (2.06E+01)	7.34E+02 (2.59E-02)	1.55E+06 (5.46E+01)	7.03E+05 (2.48E+01)	2.83E+06
Pu239	2.92E+04 (1.77E+00)	1.50E+03 (9.08E-02)	1.15E+06 (6.96E+01)	4.71E+05 (2.85E+01)	1.65E+06
U236	4.87E+04 (3.87E+00)	1.34E+02 (1.07E-02)	8.58E+05 (6.82E+01)	3.52E+05 (2.79E+01)	1.26E+06
Zr93	6.72E+03 (9.47E-01)	1.41E+04 (1.99E+00)	4.62E+05 (6.52E+01)	2.26E+05 (3.19E+01)	7.09E+05
Pu240	5.74E+03 (8.76E-01)	2.83E+02 (4.33E-02)	4.57E+05 (6.98E+01)	1.92E+05 (2.93E+01)	6.55E+05
Tc99	4.40E+03 (7.62E-01)	9.24E+03 (1.60E+00)	3.96E+05 (6.86E+01)	1.68E+05 (2.91E+01)	5.78E+05
Cs137	2.88E+03 (7.11E-01)	5.21E+03 (1.29E+00)	2.80E+05 (6.92E+01)	1.17E+05 (2.88E+01)	4.05E+05
Th232 ‡	3.42E+05 (9.46E+01)	1.94E+04 (5.37E+00)	5.70E-01 (1.58E-04)	2.36E-01 (6.52E-05)	3.62E+05
Sr90	2.36E+03 (8.80E-01)	3.49E+03 (1.30E+00)	1.87E+05 (6.95E+01)	7.62E+04 (2.84E+01)	2.69E+05

Table C.1-7. YMP-Scale Source Term Mole Inventory (Continued) (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory								
	DOE-Owned				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[mole]	[%]	[mole]	[%]	[mole]	[%]	[mole]	[%]	[mole]
Top 10 Radionuclides (Ranked on Inventory for DOE-Owned Wastes Only)									
U238	1.09E+07	(4.06E+00)	2.22E+05	(8.29E-02)	--	--	--	--	1.11E+07
U235	5.83E+05	(2.06E+01)	7.34E+02	(2.59E-02)	--	--	--	--	5.84E+05
Th232 ‡	3.42E+05	(9.46E+01)	1.94E+04	(5.37E+00)	--	--	--	--	3.61E+05
U236	4.87E+04	(3.87E+00)	1.34E+02	(1.07E-02)	--	--	--	--	4.88E+04
Pu239	2.92E+04	(1.77E+00)	1.50E+03	(9.08E-02)	--	--	--	--	3.07E+04
Zr93	6.72E+03	(9.47E-01)	1.41E+04	(1.99E+00)	--	--	--	--	2.08E+04
Tc99	4.40E+03	(7.62E-01)	9.24E+03	(1.60E+00)	--	--	--	--	1.36E+04
Cs137	2.88E+03	(7.11E-01)	5.21E+03	(1.29E+00)	--	--	--	--	8.09E+03
Pu240	5.74E+03	(8.76E-01)	2.83E+02	(4.33E-02)	--	--	--	--	6.02E+03
Sr90	2.36E+03	(8.80E-01)	3.49E+03	(1.30E+00)	--	--	--	--	5.85E+03
Top 10 Radionuclides (Ranked on Inventory for Commercial SNFs Only)									
U238	--	--	--	--	1.69E+08	(6.28E+01)	8.86E+07	(3.30E+01)	2.58E+08
U235	--	--	--	--	1.55E+06	(5.46E+01)	7.03E+05	(2.48E+01)	2.25E+06
Pu239	--	--	--	--	1.15E+06	(6.96E+01)	4.71E+05	(2.85E+01)	1.62E+06
U236	--	--	--	--	8.58E+05	(6.82E+01)	3.52E+05	(2.79E+01)	1.21E+06
Zr93	--	--	--	--	4.62E+05	(6.52E+01)	2.26E+05	(3.19E+01)	6.88E+05
Pu240	--	--	--	--	4.57E+05	(6.98E+01)	1.92E+05	(2.93E+01)	6.49E+05
Tc99	--	--	--	--	3.96E+05	(6.86E+01)	1.68E+05	(2.91E+01)	5.64E+05
Cs137	--	--	--	--	2.80E+05	(6.92E+01)	1.17E+05	(2.88E+01)	3.97E+05
Sr90	--	--	--	--	1.87E+05	(6.95E+01)	7.62E+04	(2.84E+01)	2.63E+05
Am241	--	--	--	--	1.72E+05	(6.79E+01)	7.96E+04	(3.14E+01)	2.52E+05

† Data values for radionuclides were previously reported in 1993 TSPA (Ref. TSPA 1993).

‡ Data values for radionuclides were not previously reported in 1993 TSPA (Ref. TSPA 1993).

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997). (In total 41 radionuclides are inventoried in the INEEL/PA-DB).

(b) Mole inventory values calculated using half-lives from the Decay Libraries from ORIGEN2 (Ref. Croff 1980).

(c) Note, the total DOE-owned mole load (due to radionuclides) is only 1.22E+07 moles. Thus only 4.39 % of the total mole load (due to radionuclides) in YMP is due to DOE-Owned inventory.

(d) Note, the total commercial mole load (due to radionuclides) is 2.66E+08 mole. Thus 95.6 % of the total mole load (due to radionuclides) in YMP is due to commercial inventory.

Table C.1-8. Fissile Equivalent of Many of the Radionuclides in the Actinide Series and Fission Yield Products

Nuclide		Fissile Gram Equivalent (FGE)				Fission Yield Products (d) [%]
		U-235 FGE (a)			Pu-239 FGE (b)	
ID	Atomic Number	Water-Reflected	Steel-Reflected	Conservative Value (c)	Value [%]	
Se79	34	--	--	--	--	4.50E-02
Zr93	40	--	--	--	--	6.35E+00
Nb93m	41	--	--	--	--	6.35E+00
Nb94	41	--	--	--	--	6.47E+00
Pd107	46	--	--	--	--	1.46E-01
I129	53	--	--	--	--	0.54E+00
Cs135	55	--	--	--	--	6.54E+00
Cs137	55	--	--	--	--	6.19E+00
Sm151	62	--	--	--	--	4.19E-01
Ac227	89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Th228	90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Th230	90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Th232	90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Pa231	91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
U232	92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
U233	92	1.00E+00	--	2.00E+00 (e)	1.00E+00	--
U234	92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
U235	92	1.00E+00	--	1.00E+00	1.00E+00	--
U236	92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
U237	92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
U238	92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Np237	93	1.50E-02	2.25E-02	2.25E-02 (f)	1.50E-02	--
Pu236	94	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Pu238	94	1.13E-01	1.50E-01	1.50E-01	1.13E-01	--
Pu239	94	1.00E+00	--	2.00E+00 (e)	1.00E+00	--
Pu240	94	2.25E-02	3.00E-02	3.00E-02	2.25E-02	--
Pu241	94	2.25E+00	--	2.25E-02	2.25E+00	--
Pu242	94	7.50E-03	1.13E-02	1.13E-02	7.50E-03	--
Pu244	94	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Am241	95	1.88E-02	2.81E-02	2.81E-02	1.87E-02	--
Am242m	95	3.46E+01	--	3.46E+01	3.46E+01	--
Am243	95	1.29E-02	1.80E-02	1.80E-02	1.29E-02	--
Cm242	96	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Cm243	96	5.00E+00	--	5.00E+00	5.00E+00	--
Cm244	96	9.00E-02	1.50E-01	1.50E-01	9.00E-02	--
Cm245	96	1.50E-01	--	1.50E-01	1.50E+01	--
Cm246	96	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Cm247	96	5.00E-01	--	5.00E-01	5.00E-01	--
Cm248	96	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Cm250	96	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Bk247	97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Bk249	97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--

Table C.1-8. Fissile Equivalent of Many of the Radionuclides in the Actinide Series and Fission Yield Products (Continued)

Nuclide		Fissile Gram Equivalent (FGE)				Fission Yield Products (d) [%]
		U-235 FGE (a)			Pu-239 FGE (b)	
ID	Atomic Number	Water-Reflected	Steel-Reflected	Conservative Value (c)	Value [%]	
Cf249	98	4.50E+01	--	4.50E+01	4.50E+01	--
Cf250	98	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Cf251	98	9.00E+01	--	9.00E+01	9.00E+01	--
Cf252	98	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Es252	99	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--
Es254	99	0.00E+00	0.00E+00	0.00E+00	0.00E+00	--

-- Data values not reported in references.

(a) Data taken from Ref. Rechard 1995b (Vol3/Table H-19/pg. H-17). These FGE values are based on subcritical mass limits for water-reflected or steel-reflected fissile actinide nuclides. Based on data in ANSI/ANS-8.15-1981. Zero valued entries reflect isotopes that are short-lived and/or have small fission cross-sections.

(b) Data taken from Ref. DOE 1989 (TRUPACT-II SAR/Table 10.1/pg. 1.3.7-51). Data originally from (Refs. 10.2.1 and 10.2.2 from DOE 1989).

(c) These conservative values (for U-235 FGEs only) are the maximum of water-reflected and steel-reflected derived FGEs.

(d) Fission yield products presented here are for isobars, see discussion from footnote (a) of Table C.1-11.

(e) U-233 and Pu-239 are typically given a FGE value (when expressed in units of U-235 FGE) of 2.0 for criticality safety analysis, since homogeneous aqueous slurries required about half the fissile material than that for U-235.

(f) Radionuclides which have even numbers of neutrons have reduced FGE values if the reflector is also a moderator (e.g., water), that is these radionuclides have neutron energy thresholds, below which the probability of fission is greatly reduced.

Table C.1-9. YMP-Scale Source Term Fissile Inventory (U-235 Fissile Equivalents, Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Fissile Inventory (b)				
	DOE-Owned		Commercial		Total
	SNF	DHLW	PWR	BWR	
	[kg] [%]	[kg] [%]	[kg] [%]	[kg] [%]	[kg]
Ac227	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Ag108m †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Am241	8.68E+00 (5.07E-01)	3.33E+00 (1.94E-01)	1.16E+03 (6.79E+01)	5.39E+02 (3.14E+01)	1.71E+03
Am242m	2.86E+00 (5.68E-02)	3.20E-01 (6.35E-03)	3.34E+03 (6.62E+01)	1.70E+03 (3.37E+01)	5.04E+03
Am243	1.63E-01 (1.08E-01)	1.56E-02 (1.03E-02)	1.06E+02 (6.96E+01)	4.60E+01 (3.03E+01)	1.52E+02
C14	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Cl36	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Cm243 †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Cm244	1.43E-01 (6.75E-02)	9.94E-02 (4.69E-02)	1.52E+02 (7.16E+01)	5.99E+01 (2.83E+01)	2.12E+02
Cm245	2.66E-03 (1.34E-02)	2.32E-04 (1.16E-03)	1.47E+01 (7.38E+01)	5.20E+00 (2.62E+01)	1.99E+01
Cm246	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Cs135	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Cs137	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
I129	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Mo93 †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Nb93m ‡	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Nb94	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Ni59	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Ni63	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Np237	5.17E+00 (5.55E-01)	4.25E+00 (4.55E-01)	6.77E+02 (7.26E+01)	2.46E+02 (2.64E+01)	9.33E+02
Pa231	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Pb210	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Pd107	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Pu238	5.26E+00 (2.65E-01)	1.65E+01 (8.29E-01)	1.44E+03 (7.23E+01)	5.28E+02 (2.66E+01)	1.99E+03
Pu239	1.40E+04 (1.77E+00)	7.18E+02 (9.08E-02)	5.50E+05 (6.96E+01)	2.25E+05 (2.85E+01)	7.90E+05
Pu240	4.14E+01 (8.76E-01)	2.04E+00 (4.33E-02)	3.29E+03 (6.98E+01)	1.38E+03 (2.93E+01)	4.72E+03
Pu241	1.35E+00 (1.67E-01)	1.52E-01 (1.88E-02)	5.53E+02 (6.83E+01)	2.56E+02 (3.16E+01)	8.10E+02
Pu242	7.61E-01 (1.90E-01)	6.99E-02 (1.74E-02)	2.73E+02 (6.81E+01)	1.27E+02 (3.17E+01)	4.00E+02
Ra226	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Ra228 ‡	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Se79	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Sm151	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Sn121m †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Sn126	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Sr90	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Tc99	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Th229	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Th230	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Th232 ‡	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00

Table C.1-9. YMP-Scale Source Term Fissile Inventory (Continued)
(U-235 Fissile Equivalents, Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Fissile Inventory (b)				
	DOE-Owned		Commercial		Total
	SNF	DHLW	PWR	BWR	
	[kg] [%]	[kg] [%]	[kg] [%]	[kg] [%]	[kg]
U232 †	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
U233	2.48E+03 (9.99E+01)	5.68E-01 (2.29E-02)	5.31E-01 (2.14E-02)	1.91E-01 (7.71E-03)	2.48E+03
U234	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
U235	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	6.66E+05
U236	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
U238	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Zr93	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00 (0.00E+00)	0.00E+00
Total	1.53E+05 (1.04E+01)	9.18E+02 (6.22E-02)	9.25E+05 (6.27E+01)	3.95E+05 (2.68E+01)	1.47E+06
	1.54E+05 (10.5 %) (c)		1.32E+06 (89.5 %) (d)		
Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned and Commercial)					
Pu239	1.40E+04 (1.77E+00)	7.18E+02 (9.08E-02)	5.50E+05 (6.96E+01)	2.25E+05 (2.85E+01)	7.90E+05
U235	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	6.66E+05
Am242m	2.86E+00 (5.68E-02)	3.20E-01 (6.35E-03)	3.34E+03 (6.62E+01)	1.70E+03 (3.37E+01)	5.04E+03
Pu240	4.14E+01 (8.76E-01)	2.04E+00 (4.33E-02)	3.29E+03 (6.98E+01)	1.38E+03 (2.93E+01)	4.72E+03
U233	2.48E+03 (9.99E+01)	5.68E-01 (2.29E-02)	5.31E-01 (2.14E-02)	1.91E-01 (7.71E-03)	2.48E+03
Am241	8.68E+00 (5.07E-01)	3.33E+00 (1.94E-01)	1.16E+03 (6.79E+01)	5.39E+02 (3.14E+01)	1.71E+03
Np237	5.17E+00 (5.55E-01)	4.25E+00 (4.55E-01)	6.77E+02 (7.26E+01)	2.46E+02 (2.64E+01)	9.33E+02
Pu241	1.35E+00 (1.67E-01)	1.52E-01 (1.88E-02)	5.53E+02 (6.83E+01)	2.56E+02 (3.16E+01)	8.10E+02
Pu242	7.61E-01 (1.90E-01)	6.99E-02 (1.74E-02)	2.73E+02 (6.81E+01)	1.27E+02 (3.17E+01)	4.00E+02
Pu238	5.26E+00 (2.65E-01)	1.65E+01 (8.29E-01)	1.44E+03 (7.23E+01)	5.28E+02 (2.66E+01)	1.99E+03

† Data values for radionuclides were previously reported in 1993 TSPA (Ref. TSPA 1993).
‡ Data values for radionuclides were not previously reported in 1993 TSPA (Ref. TSPA 1993).

Table C.1-9. YMP-Scale Source Term Fissile Inventory (Continued)
(U-235 Fissile Equivalents, Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Inventory				
	DOE-Owned		Commercial		Total
	SNF	DHLW	PWR	BWR	
	[kg] [%]	[kg] [%]	[kg] [%]	[kg] [%]	[kg]
Top 10 Radionuclides (Ranked on Total Inventory for DOE-Owned Wastes Only)					
U235	1.37E+05 (2.06E+01)	1.73E+02 (2.59E-02)	– –	– –	1.37E+05
Pu239	1.40E+04 (1.77E+00)	7.18E+02 (9.08E-02)	– –	– –	1.47E+04
U233	2.48E+03 (9.99E+01)	5.68E-01 (2.29E-02)	– –	– –	2.48E+03
Pu240	4.14E+01 (8.76E-01)	2.04E+00 (4.33E-02)	– –	– –	4.34E+01
Am241	8.68E+00 (5.07E-01)	3.33E+00 (1.94E-01)	– –	– –	1.20E+01
Np237	5.17E+00 (5.55E-01)	4.25E+00 (4.55E-01)	– –	– –	9.42E+00
Pu238	5.26E+00 (2.65E-01)	1.65E+01 (8.29E-01)	– –	– –	2.18E+01
Am242m	2.86E+00 (5.68E-02)	3.20E-01 (6.35E-03)	– –	– –	3.18E+00
Pu241	1.35E+00 (1.67E-01)	1.52E-01 (1.88E-02)	– –	– –	1.50E+00
Pu242	7.61E-01 (1.90E-01)	6.99E-02 (1.74E-02)	– –	– –	8.31E-01
Top 10 Radionuclides (Ranked on Total Inventory for Commercial SNFs Only)					
Pu239	– –	– –	5.50E+05 (6.96E+01)	2.25E+05 (2.85E+01)	7.75E+05
U235	– –	– –	3.64E+05 (5.46E+01)	1.65E+05 (2.48E+01)	5.29E+05
Am242m	– –	– –	3.34E+03 (6.62E+01)	1.70E+03 (3.37E+01)	5.04E+03
Pu240	– –	– –	3.29E+03 (6.98E+01)	1.38E+03 (2.93E+01)	4.67E+03
Am241	– –	– –	1.16E+03 (6.79E+01)	5.39E+02 (3.14E+01)	1.70E+03
Np237	– –	– –	6.77E+02 (7.26E+01)	2.46E+02 (2.64E+01)	9.23E+02
Pu241	– –	– –	5.53E+02 (6.83E+01)	2.56E+02 (3.16E+01)	5.85E+02
Pu242	– –	– –	2.73E+02 (6.81E+01)	1.27E+02 (3.17E+01)	4.00E+02
Pu238	– –	– –	1.44E+03 (7.23E+01)	5.28E+02 (2.66E+01)	1.97E+03
Cm244	– –	– –	1.52E+02 (7.16E+01)	5.99E+01 (2.83E+01)	2.12E+02

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997). (In total 41 radionuclides are inventoried in the INEEL/PA-DB).

(b) Fissile inventory values calculated using mass values form Table C.1-7 and U-235 fissile gram equivalents (FGEs) from Table C.1-8.

(c) Note, the total DOE-owned FGE load (due to radionuclides) is only 1.54E+05 kg. Thus only 10.5 % of the total FGE load (due to radionuclides) in YMP is due to DOE-Owned inventory.

(d) Note, the total commercial FGE load (due to radionuclides) is 1.32E+06 kg. Thus 89.5 % of the total FGE load (due to radionuclides) in YMP is due to commercial inventory.

Table C.1-10. YMP-Scale Source Term Fissile Enrichments (Calendar Year = 2035 , Time = 0 yr) (a)

Fissile Material		YMP Radionuclide Fissile Inventory				
		DOE-Owned		Commercial		Total
ID	Property	SNF	DHLW	PWR	BWR	
Uranium (U232,U233, U234,U235, U236 & U238)	FGE [kg] (b) Mass [kg] (c) Enrichment	1.39E+05 2.75E+06 5.05 (%)	1.74E+02 5.32E+04 0.327 (%)	3.64E+05 4.08E+07 0.893 (%)	1.65E+05 2.14E+07 0.773 (%)	6.68E+05 6.49E+07
	FGE [kg] Mass [kg] Enrichment	1.40E+05 2.80E+06 4.99 (%) (d)		5.29E+05 6.21E+07 **0.852 (%) (e)**		
	FGE [kg] Mass [kg] Enrichment	6.69E+05 6.49E+07 **YMP-Site Average = 1.03 (%) (f)**				
Plutonium (Pu238, Pu239, Pu240, Pu241, & Pu242)	FGE [kg] (b) Mass [kg] (c) Enrichment	7.15E+03 8.52E+03 83.9 (%)	3.88E+02 5.50E+02 70.8 (%)	3.34E+05 4.43E+05 75.4 (%)	1.40E+05 1.85E+05 75.7 (%)	4.81E+05 6.38E+05
	FGE [kg] Mass [kg] Enrichment	7.54E+03 9.07E+03 83.1 (%) (d)		4.74E+05 6.29E+05 **75.4 (%) (e)**		
	FGE [kg] Mass [kg] Enrichment	4.81E+05 6.38E+05 **YMP-Site Average = 75.5 (%) (f)**				

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997).

(b) Fissile mass inventory values taken from Table C.1-9.

(c) Mass inventory values taken from Table C.1-6.

(d) YMP-Site average enrichment for DOE-Owned inventory.

Table C.1-10. YMP-Scale Source Term Fissile Enrichments (Continued) (Calendar Year = 2035 , Time = 0 yr) (a)

Fissile Material		YMP Radionuclide Fissile Inventory				
		DOE-Owned		Commercial		Total
ID	Property	SNF	DHLW	PWR	BWR	
Uranium + Plutonium (U233, U234, U235, U236, U238, Pu238, Pu239, Pu240, Pu241 + Pu242)	FGE [kg] (b) Mass [kg] (c) Enrichment	1.53E+05 2.76E+06 5.54 (%)	9.11E+02 5.38E+04 1.69 (%)	9.20E+05 4.12E+07 2.23 (%)	3.92E+05 2.15E+07 1.82 (%)	1.46E+06 6.56E+07
	FGE [kg] Mass [kg] Enrichment	1.54E+05 2.81E+06 5.48 (%) (d)		1.31E+06 6.28E+07 **2.09 (%) (e)**		
	FGE [kg] Mass [kg] Enrichment	1.46E+06 6.56E+07 **YMP-Site Average = 2.23 (%) (f)**				
Uranium + Plutonium + others (g) (U233, U234, U235, U236, U238, Pu238, Pu239, Pu240, Pu241, Pu242, Am241, Am242m, Am243, Cm244 & Np237)	FGE [kg] (b) Mass [kg] (c) Enrichment	1.53E+05 2.76E+06 5.54 (%)	9.18E+02 5.38E+04 1.71 (%)	9.25E+05 4.12E+07 2.25 (%)	3.95E+05 2.15E+07 1.84 (%)	1.47E+06 6.56E+07
	FGE [kg] Mass [kg] Enrichment	1.54E+05 2.81E+06 5.48 (%) (d)		1.32E+06 6.28E+07 **2.10 (%) (e)**		
	FGE [kg] Mass [kg] Enrichment	1.47E+06 6.56E+07 **YMP-Site Average = 2.24 (%) (f)**				

(e) YMP-Site average enrichment for commercial SNF.

(f) Comparison of this average value to that for commercial SNFs shows that the impact of additional fissile material from DOE-owned inventory is relatively small. If necessary, depleted uranium can be added to DOE-own inventory to reduce enrichment to commercial values (2.10%) and not impact YMP-Site values.

(g) This special case used all FGE inventories (i.e., including Am241, Am242m ...) and identifies that the non-(uranium and plutonium) fissile nuclides do not significantly impact the YMP-Scale enrichment.

(h) Fissile mass inventory values were not taken from Table C.1-9 (i.e., U-235 FGE values only) since this case corresponds only to plutonium nuclides. Thus Pu-239 fissile mass values were calculated for this case using Table C.1-8 Pu-239 FGE constants. Cases that combine effects of uranium and plutonium are presented in U-235 FGEs only since uranium fissile nuclides are dominant.

Table C.1-11. YMP-Scale Source Term Fission Load (Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Fission Load (b)								
	DOE-Owned (c)				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[fission]	[%]	[fission]	[%]	[fission]	[%]	[fission]	[%]	[fission]
Ac227	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Am241	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Am242m	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Am243	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Ag108m †	--	(--)	--	(--)	--	(--)	--	(--)	--
C14	NC	(--)	--	(--)	NC	(--)	NC	(--)	--
Cl36	NC	(--)	--	(--)	NC	(--)	NC	(--)	--
Cm243 †	--	(--)	--	(--)	--	(--)	--	(--)	--
Cm244	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Cm245	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Cm246	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Cs135	1.21E+28	(5.74E-01)	3.20E+28	(1.52E+00)	1.42E+30	(6.72E+01)	6.46E+29	(3.07E+01)	2.11E+30
Cs137	2.80E+28	(7.11E-01)	5.07E+28	(1.29E+00)	2.73E+30	(6.92E+01)	1.13E+30	(2.88E+01)	3.94E+30
I129	5.78E+28	(7.02E-01)	3.15E+25	(3.83E-04)	5.75E+30	(6.98E+01)	2.43E+30	(2.95E+01)	8.23E+30
Mo93 †	--	(--)	--	(--)	--	(--)	--	(--)	--
Nb93m ‡	1.19E+19	(5.37E-01)	5.37E+19	(2.43E+00)	1.43E+21	(6.50E+01)	7.08E+20	(3.21E+01)	2.21E+21
Nb94	5.44E+24	(1.87E-02)	7.36E+22	(2.53E-04)	2.80E+28	(9.63E+01)	1.08E+27	(3.71E+00)	2.91E+28
Ni59	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Ni63	NC	(--)	--	(--)	NC	(--)	NC	(--)	--
Np237	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pa231	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pb210	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pd107	2.14E+29	(3.33E-01)	0.00E+00	(0.00E+00)	4.48E+31	(6.96E+01)	1.93E+31	(3.01E+01)	6.43E+31
Pu238	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pu239	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pu240	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pu241	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Pu242	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Ra226	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Ra228 ‡	NC	(--)	--	(--)	NC	(--)	NC	(--)	--
Se79	5.89E+28	(7.77E-01)	1.07E+29	(1.42E+00)	5.25E+30	(6.91E+01)	2.18E+30	(2.87E+01)	7.59E+30
Sm151	1.84E+28	(1.93E+00)	0.00E+00	(0.00E+00)	6.58E+29	(6.88E+01)	2.81E+29	(2.93E+01)	9.57E+29
Sn121m †	--	(--)	--	(--)	--	(--)	--	(--)	--
Sn126	8.99E+28	(5.46E-01)	0.00E+00	(0.00E+00)	1.16E+31	(7.02E+01)	4.83E+30	(2.93E+01)	1.65E+31
Sr90	2.45E+28	(8.80E-01)	3.62E+28	(1.30E+00)	1.94E+30	(6.95E+01)	7.91E+29	(2.84E+01)	2.79E+30
Tc99	4.35E+28	(7.62E-01)	9.12E+28	(1.60E+00)	3.91E+30	(6.86E+01)	1.66E+30	(2.91E+01)	5.71E+30
Th229	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Th230	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Th232	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC

Table C.1-11. YMP-Scale Source Term Fission Load (Continued)
(Calendar Year = 2035 , Time = 0 yr) (a)

Nuclide ID	YMP Radionuclide Fission Load (b)								
	DOE-Owned (c)				Commercial				Total
	SNF		DHLW		PWR		BWR		
	[fission]	[%]	[fission]	[%]	[fission]	[%]	[fission]	[%]	[fission]
U232 †	--	(--)	--	(--)	--	(--)	--	(--)	--
U233	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
U234	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
U235	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
U236	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
U238	NC	(--)	NC	(--)	NC	(--)	NC	(--)	NC
Zr93	6.35E+28	(9.47E-01)	1.34E+29	(1.99E+00)	4.37E+30	(6.52E+01)	2.14E+30	(3.19E+01)	6.71E+30
Maximum Pd107	2.14E+29	(3.33E-01)	0.00E+00	(0.00E+00)	4.48E+31	(6.96E+01)	1.93E+31	(3.01E+01)	6.43E+31

† Data values for radionuclides were previously reported in 1993 TSPA (Ref. TSPA 1993).

‡ Data values for radionuclides were not previously reported in 1993 TSPA (Ref. TSPA 1993).

NC,-- Data not calculated since radionuclide does not fission.

(a) Spent nuclear fuel (SNF) and high-level waste (HLW) inventory data taken from INEEL/PA 1997 Parameters Database (values represent intermediate database values, upgraded values can be found in Appendix A of Ref. INEEL 1997). (In total 41 radionuclides are inventoried in the INEEL/PA-DB). The total number of fissions are calculated using the expression: FISSIONS = (MOLES x Na / FYP), where the mole values are taken from Table C.1-7, Na is Avogadro's constant, and FYP are the fission yield products taken from the Chart of the Nuclides (Ref. GE 1989). These calculations, most interpretable by nuclear engineers, indicate an estimate of the number of nuclear fissions in the spent nuclear fuel to be placed in the YMP site. As can be noted, the estimated number of total fissions produced with information from the various radionuclides vary significantly. This results since: 1) the fission yield products used are intended for isobars, thus using them for individual nuclides would result in underestimations of total fissions, 2) if the radionuclide being investigated has a short half-life, then substantial radioactive decay is incorporated into the mole inventory values (from Table C.1-7), and 3) if the radionuclide is not near the nuclear stability diagonal, then the effects of items 1 and 2 may both be significant. Thus the largest estimated "total fission" value would be the most appropriate (i.e., the value of ~ 6.E+31 as identified from Pd107). Also noticeable from this table is that on the order of 1 percent of the total inventory of fissions is due to DOE-owned wastes, this is consistent with Table C.1-3.

(b) Fission inventory values calculated using radionuclide inventory values from Table C.1-7 and fission yield products from the Chart of the Nuclides (see Table C.1-8, Ref. GE-1).

(c) Note, the total DOE-owned fission load (due to radionuclides) is on the order of only one percent (i.e., on the order of 99 percent of the fissions are contributed due to Commercial SNFs.

APPENDIX D

Identification of Release of Radionuclides to Biosphere

D.1 Introduction

Environmental regulations for the proposed Yucca Mountain repository will be identified in 40 CFR 197, which is expected to be promulgated in the near future. This compliance standard is expected to be based on health effects in human beings due to exposure from radionuclides released to the biosphere at the accessible environment. The 1997 INEEL PA primarily used criteria in 40 CFR 191 Subpart B, specifically the 5 km exclusion zone and the individual protection requirements. The 5 km boundary is expected to change (farming activities are currently done in the Armagosa Valley at approximately 20-25 km from the repository site), as are the regulations for individual protection requirements (for times beyond 10,000 years and limiting dose). In previous PAs, the most significant performance metric was cumulative radioactive releases (in EPA units, see Appendix B). However, more recent guidance from the National Academy of Science (NAS/NRC, 1995) indicates that criteria similar to the Individual Protection Requirement of 40 CFR 191 Subpart B, referred to above, should now take precedence. Regardless of the criteria used, releases to the biosphere remain a significant measure of performance assessment.

D.2 Dose Results From Previous Performance Assessments

The measure of health affects used in the previous PAs has been the annual effective dose equivalent (AEDE, referred to as "the dose") received by maximally exposed individuals for the exposure pathways that have been historically considered. The dose received by a maximally exposed individual has been found to be due to exposure pathways for radionuclides transported to the accessible environment via groundwater. Specifically, farming activities (including drinking water and irrigation), using a water-well drilled into a contaminated aquifer, have been shown to be the exposure scenario which yields the largest doses to human beings. The radionuclides which contribute significantly to the dose depend on various factors; among those being travel time and distance to the accessible environment, infiltration rates, package failure rates, solubilities and retardation (nuclide sorption).

The 1994 and 1997 INEEL PA (Rechard, 1995b; 1997) have shown that ^{99}Tc and ^{129}I are the dominate contributors to dose for approximately the first 10,000 to 50,000 years; and that from 50,000 to sometime beyond 100,000, ^{237}Np, becomes the dominate contributor. Figure D.2-1 shows the 1994 INEEL PA results for water-borne dose from horizontal and vertical emplacement for zero to 100,000 years at 1,000 meters downstream from the repository (farm family pathway). Figure D.2-2 shows the 1997 results at the 5 km boundary for zero to 100,000 years including the total mean dose and the contributions from ^{99}Tc, ^{129}I, and ^{237}Np. An important caveat in the 1997 INEEL PA is that only three radionuclides were transported in the unsaturated and saturated zones (^{99}Tc, ^{129}I and ^{237}Np) to the accessible environment. This was motivated by the earlier 1994 INEEL PA results which indicated those nuclides to be the most significant contributors to dose and due to the fact that there was limited computational resources to transport more nuclides. It should be noted that the 1994 INEL PA considered DOE-owned SNF and HLW inventories only, the 1997 INEEL PA also considered commercial SNF inventories which are comprised of PWR and BWR SNF.

The previous PA analyses have not considered dose for times beyond 100,000 years. However, it is expected that ^{237}Np contribution may continue to increase beyond 100,000 years. The 1994 PA results also indicate that the doses due to ^{231}Pa, ^{233}U, and ^{234}U and daughter products such as ^{226}Ra or ^{210}Pb, (though modest in comparison to doses due to ^{99}Tc, ^{129}I, and ^{237}Np, previous to 100,000 years) may be an important group of nuclides for potential health effects beyond 100,000 years. Plots of the activity time histories (from the 1997 INEEL PA) for ^{233}U, ^{231}Pa, and ^{226}Ra are included in Figures D.2-3, D.2-4, and D.2-5.

The total activity (curies) inventory can be seen in Table C.1-3 (1997 INEEL PA Parameter Database). Table D.2-1 summarizes the biosphere exposure models and which major categories of SNFs are modeled in the available historical performance assessments for the Yucca Mountain repository.

In Table D.2-2, the total activity and mean dose contributions due to ^{99}Tc, ^{129}I, and ^{237}Np, (as calculated in the 1997 INEEL PA) for the farm family, are divided into the commercial and DOE components. The results shown assume that there is no preferential release for these two components. The corresponding time histories, from zero to 100,000 years, for the total mean dose due to all SNF categories (commingled) are in Figure D.2-2. The contributions due to the commercial and DOE SNFs are shown in Figure D.2-6. The dose is expressed as the annually effective dose equivalent (AEDE). Also shown are the dose contributions due to ^{99}Tc, ^{129}I, and ^{237}Np. The mean dose activity per package is listed in Table D.2-3 with "per package" calculations using a DOE package count of 3,131 and a commercial package count of 7,679. Figure D.2-7 presents the contributions of commercial and DOE SNFs on a per package basis and Table D.2-4 summarizes key results shown in the figure.

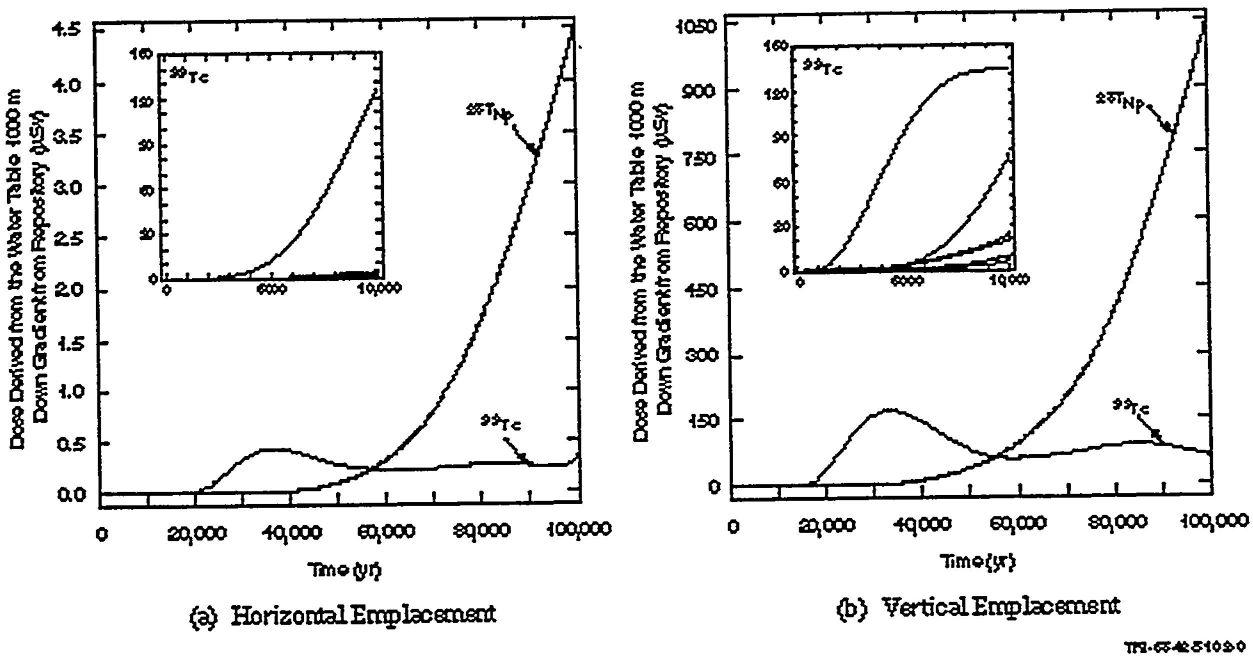


Figure D.2-1. 1994 INEEL PA results for water-born dose from horizontal and vertical emplacement for zero to 100,000 years at 1,000 meters downstream (farm family pathway).

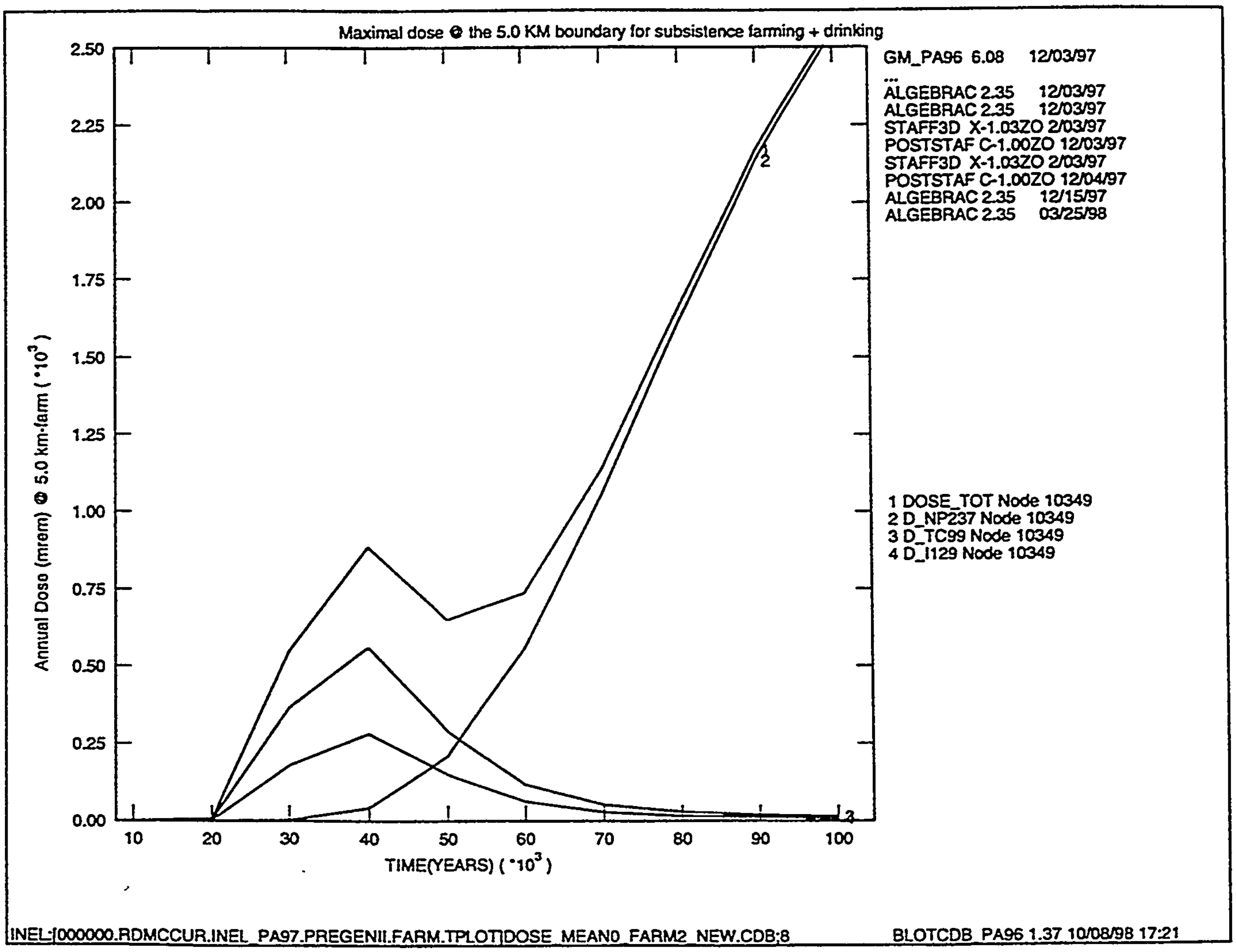


Figure D.2-2. 1997 INEEL PA results at the 5 km boundary for zero to 100,000 years including the total mean dose and contributions from ^{99}Tc, ^{129}I, and ^{237}Np.

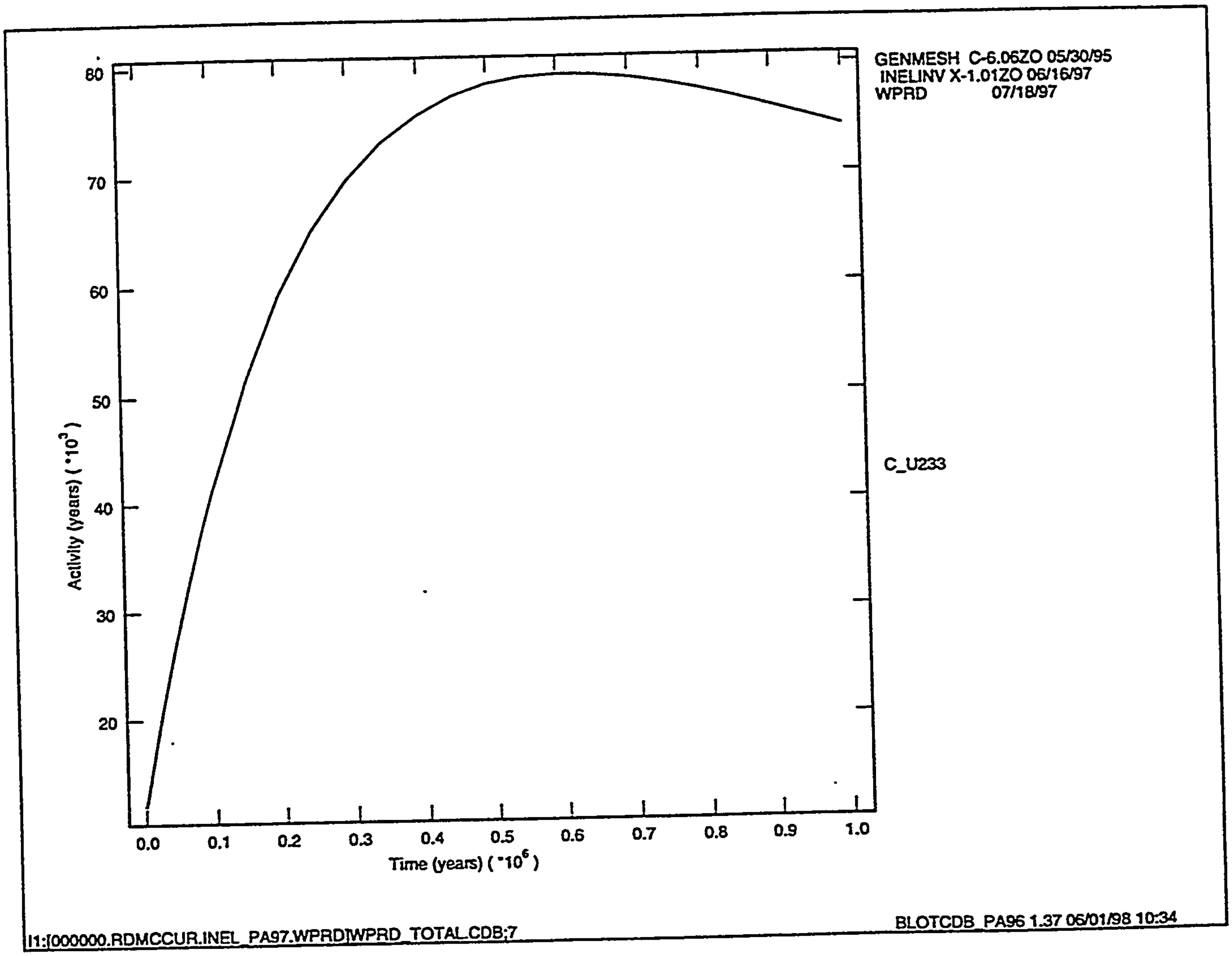


Figure D.2-3. 1997 INEEL PA results at the 5 km boundary for zero to 100,000 years including the total mean dose and contributions from ^{233}U.

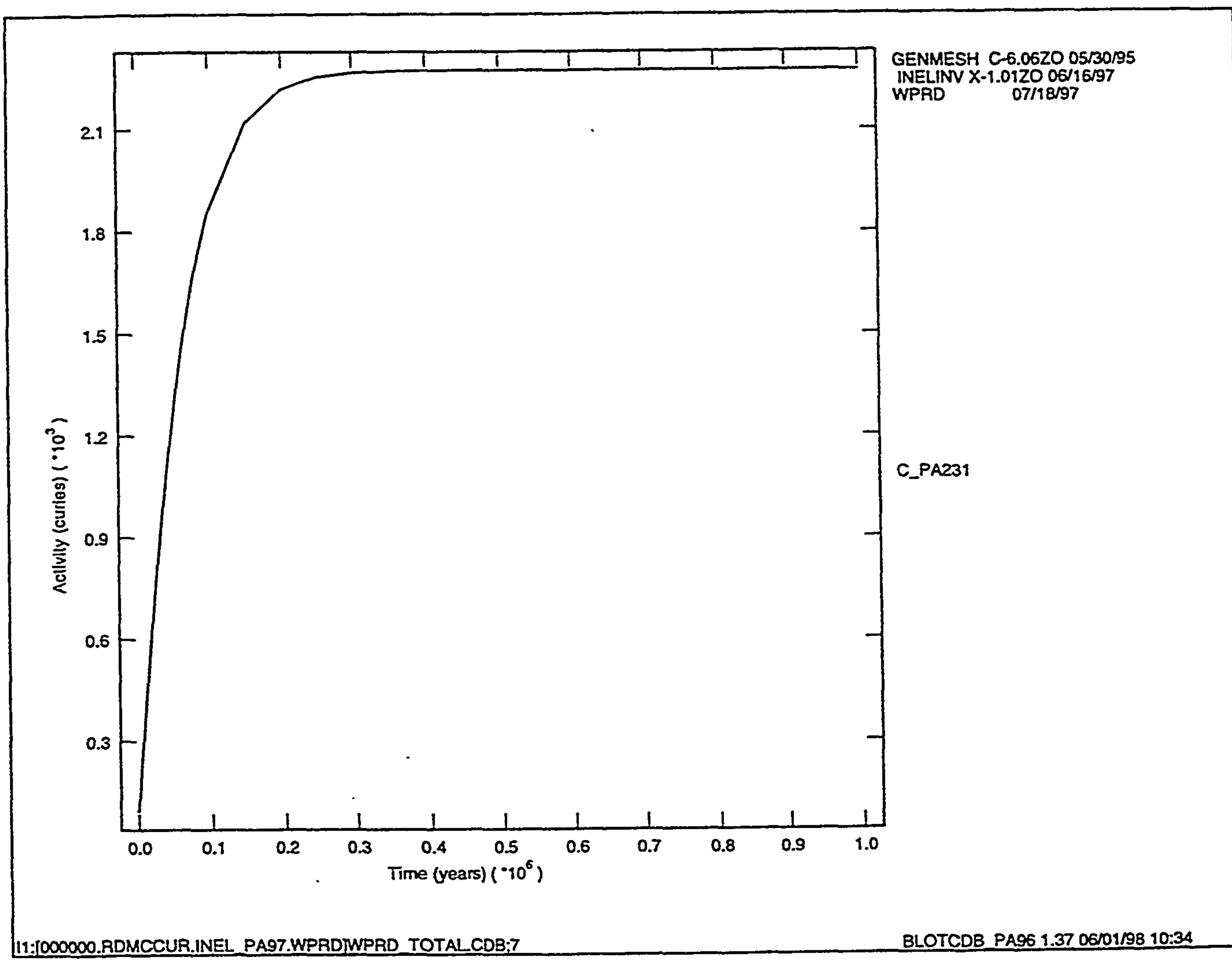


Figure D.2-4. 1997 INEEL PA results at the 5 km boundary for zero to 100,000 years including the total mean dose and contributions from ^{231}Pa.

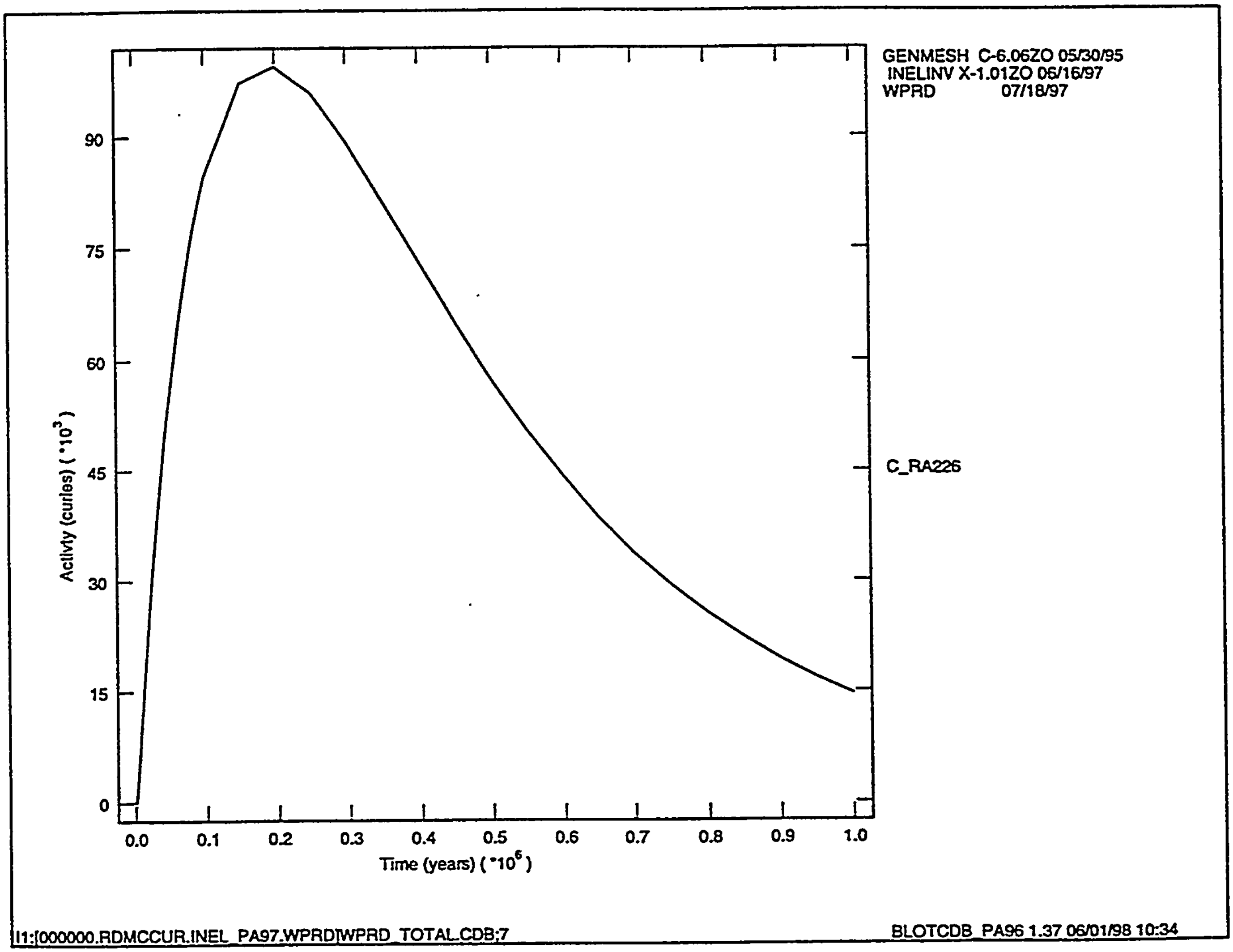


Figure D.2-5. 1997 INEEL PA results at the 5 km boundary for zero to 100,000 years including the total mean dose and contributions from ^{226}Ra.

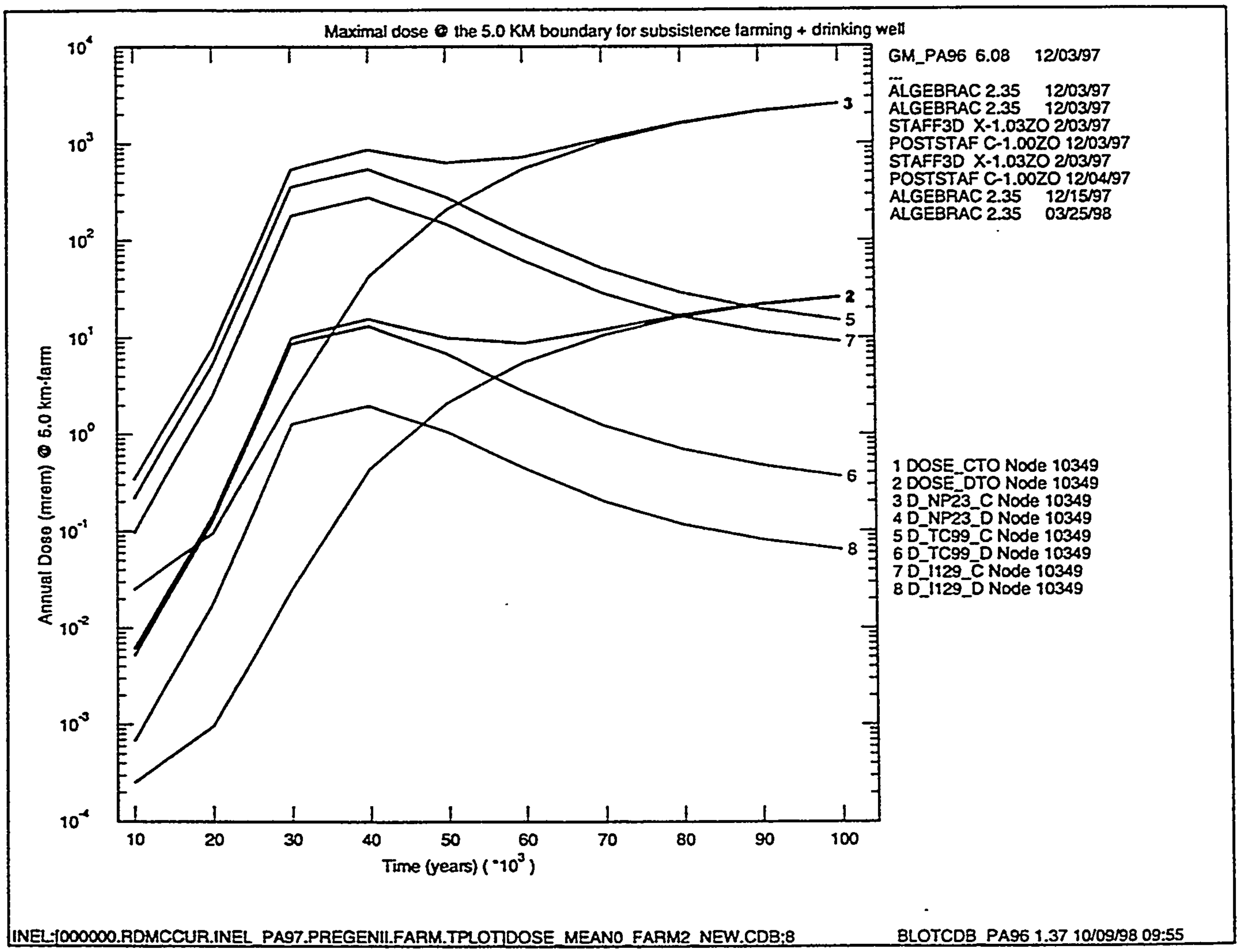


Figure D.2-6. 1997 INEEL PA results for annual effective dose equivalent (AEDE, mean total value and contributions from ^{99}Tc, ^{129}I, and ^{237}Np) at the 5 km boundary for subsistence farm family.

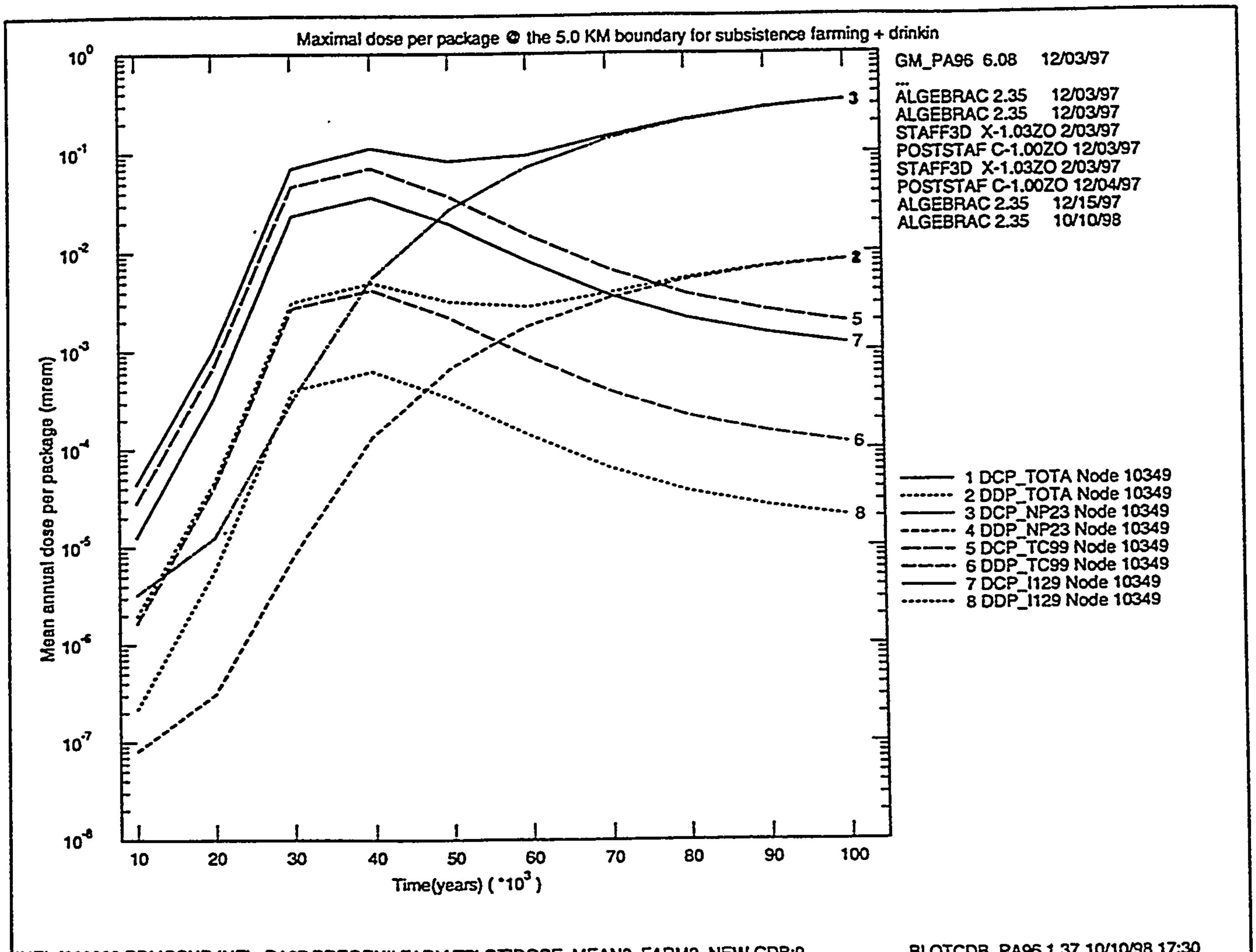


Figure D.2-7. 1997 INEEL PA results, on a per package basis, for annual effective dose equivalent (AEDE, mean total value and contributions from ^{99}Tc, ^{129}I, and ^{237}Np) at the 5 km boundary for subsistence farm family.

Table D.2-1. Fuels Key Reference Chart for Yucca Mountain Repository Biosphere Dose Calculations

	DOE	Commercial	Exposure Scenarios
1991 TSPA	X	X	Farmer, drinking water, driller (exploratory), Gardener, carbon dioxide release
1993 TSPA *	X	X	Drinking water
1994 INEEL PA **	X		Subsistence farmer, driller
1995 TSPA	X	X	Drinking Water
1997 INEEL PA	X	X	Subsistence farmer, rancher, Drinking water
1998 INEEL PA	X	X	Subsistence farm, ranch, Average resident
1997 TSPA	X	X	Farmer
1998 TSPA-VA	X	X	Subsistence farmer, residential farmer, average resident (Armagosa Valley)

* Not a full biosphere model calculation.

** 94 INEEL PA also did biosphere models for generic salt and granite repositories.

Farm family: Uses water for drinking and irrigating crops, includes inhalation of suspended particles.

Rancher: Uses water for cattle, whose meat is later consumed by rancher.

Driller: Externally exposed to drilling tailings and may have additional close-range exposure to drilling sample.

Table D.2-2 YMP-Scale Contribution to Dose (Based on 50-yr Normalized Dose).

Nuclide	Sorce Term Radionuclide Inventory			Annual Effective		
ID				Dose Equivalent		
	DOE	Comm	Total	DOE	Comm	Total
(year)	[Ci]	[Ci]	[Ci]	[mrem/yr]	[mrem/yr]	[mrem/yr]
	(%)	(%)		(%)	(%)	
Tc99	2.21E+04	9.18E+05	9.40E+05	5.24E-03	2.18E-01	2.23E-01
(10,000 yr)	(2.35)	(97.65)		(2.35)	(97.65)	
I129	1.64E+01	2.32E+03	2.34E+03	6.85E-04	9.69E-02	9.77E-02
(10,000 yr)	(0.70)	(99.30)		(0.70)	(99.30)	
Np237	9.61E+02	9.51E+04	9.61E+04	2.55E-04	2.51E-02	2.54E-02
(10,000 yr)	(1.00)	(99.00)		(1.00)	(99.00)	
Total (b)	4.89E+05	3.32E+07	3.37E+07	6.18E-03	3.40E-01	3.46E-01
	(1.45)	(98.55)		(1.70)	(98.30)	
Tc99	1.94E+04	8.06E+05	8.25E+05	6.18	2.86E+02	2.90E+02
(50,000 yr)	(2.35)	(97.65)		(2.35)	(97.65)	
I129	1.63E+01	2.31E+03	2.33E+03	1.05	1.49E+02	1.50E+02
(50,000 yr)	(0.70)	(99.30)		(0.70)	(99.30)	
Np237	9.49E+02	9.40E+04	9.49E+04	2.11	2.08E+02	2.10E+02
(50,000 yr)	(1.00)	(99.00)		(1.00)	(99.00)	
Total	1.16E+05	7.88E+06	8.00E+06	9.34	6.40E+02	6.50E+02
	(1.50)	(98.50)		(1.50)	(98.50)	
Tc99	1.65E+04	6.85E+05	7.01E+05	2.20E-01	8.98	9.2
(100,000 yr)	(2.35)	(97.65)		(2.35)	(97.65)	
I129	1.63E+01	2.31E+03	2.33E+03	1.09E-01	1.54E+01	1.55E+01
(100,000 yr)	(0.70)	(99.30)		(0.70)	(99.30)	
Np237	9.34E+02	9.25E+04	9.34E+04	2.56E+01	2.52E+03	2.55E+03
(100,000 yr)	(1.00)	(99.00)		(1.00)	(99.00)	
Total	4.74E+04	3.22E+06	3.27E+06	2.57E+01	2.54E+03	2.57E+03
	(1.00)	(99.00)		(1.00)	(99.00)	

(a) AEDE values at 5.0 km boundary, maximally exposed individual (data from preliminary analysis, Ref, Rechard 1998).
(b) Total for curies is total of all radionuclides in inventory.

Table D.2-3. Dose Due to Contents of Single Packageof DOE and Commercial SNFs

Nuclide ID	DOE * Package	Commercial Package	** DOE / Commerical	Time Post-Closure
Tc99	1.67E-06	6.94E-05	2.40E-02	10,000 Years
I129	2.19E-07	3.09E-05	7.07E-03	
Np237	8.13E-08	8.02E-06	1.01E-02	
Total	1.97E-06	1.08E-04	1.82E-02	
Tc99	2.17E-03	9.04E-02	2.41E-02	50,000 Years
I129	3.36E-04	4.75E-02	7.07E-03	
Np237	6.73E-04	6.64E-02	1.01E-02	
Total	3.18E-03	2.04E-01	1.56E-02	
Tc99	1.16E-04	4.82E-03	2.41E-02	100,000 Years
I129	2.06E-05	2.91E-03	7.07E-03	
Np237	8.16E-03	8.05E-01	1.01E-03	
Total	8.30E-03	8.12E-01	1.02E-02	

* DOE number based on HLW inventory used for 1997 INEEL PA calculations.

** Ratio of mean dose DOE package and mean dose Commercial package.

Table D.2-4 Key Results From Figure D.2-7 (data from Rechard 1998).

Curve (ID)	Fuel Type	Key Results (per package basis)
8, 6 7, 5	Commercial fuels DOE fuel	Annual doses due to ^{129}I, ^{99}Tc, (at 5.0 km boundary) dominate the total doses for the first 50->70 thousand years, peaking at approximately 40,000 years.
4 3	Commercial fuel DOE fuel	Annual dose to ^{237}Np releases continue to increase at 100,000 years with max value of about 1.0 mrem for commercial fuels and 10^{-2} mrem for DOE-owned fuels.
2 1	Commercial fuel DOE fuel	Total AEDE peaks temporarily at 40,000 years due mainly to ^{99}Tc, and ^{129}I, but increases again after 50,000 years due to ^{237}Np. Maximum values of approximately 1.0 mrem (commercial) and 10^{-2} mrem are reached at 100,000 years (limit of calculations).

APPENDIX E

Criticality Potential (CX) Model

E.1 Operation of RKeff Code

The RCS code system uses extensive automation to make parametric studies possible. Initially, a module called RKeff is used to generate input files for Monte Carlo Neutral Particle Transport code (MCNP) (Briesmeister, 1986; 1988). RKeff reads an input file created by the user called MCNP_MAS.INP, and generates the MCNP files automatically. A sample MCNP_MAS.INP file for spherical geometry is shown below (the row numbers are just for reference):

[Row]		
[1]	1	Host Rock Material (Topopah Springs Tuff)
[2]	13.9	Porosity of Host Rock Material (13.9%)
[3]	14	Fuel Type (INEEL HEU 4% U235)
[4]	11	Fissile Material Type (Pure Uranium)
[5]	sensitivity 1 35 1	Volume % of Fissile Material in Host Rock
[6]	13	Water Saturation (65%)
[7]	2	Water Composition (Pure Water)
[8]	1	Geometry (Sphere)
[9]	sensitivity 1 26 1	Radius
[10]	2	Reflector Type (Topopah Springs Tuff)
[11]	1	Type of Input File Desired (MCNP)

The word "sensitivity" is a flag for RKeff to run through all of the parameters from the beginning index number to ending index number by increments of the third number. This allows very large parametric studies to be set up quickly without having to rewrite every file by hand. The above file would write a series of MCNP files that varied the amount of fissile material in the host rock from 0.01% to 100% (or as much as allowed by the void space.) The radius sensitivity would write files for radii from 0.5 cm to 10,000 cm. Because two parameters are being varied, the combination would allow a possibility of 910 files. To help reduce this number, RKeff also calculates k_{inf}, and any file with a k_{inf} of less than 0.5 is discarded. It is still possible to examine millions of possible combinations (if enough time is available) easily. Almost all of the parameters can be varied with the sensitivity feature.

When MCNP_MAS.INP is read by RKeff, it calculates the concentration, atom fractions, and geometries for input into MCNP and writes it to a fully documented input file (shown in Appendix G). Each file is saved under a unique name that varies with the input parameters. A sample name is poakajpa.MIN (which is one of the files that was written by RKeff from the above input file and can be seen in Appendix G). Each letter of the filename corresponds to part of the input parameter as shown by the legend below:

First Character-Host Rock

b= Rock Salt
c= Pure Salt
d= Culebra
e= Clay
f= Simulated Fuel Mixture
g= Pure SiO_2
h= Pure Water
I= Graphite
p= Paintbrush Tuff 13.9%
0= Concrete
r= Rust
s= Sandstone
t= Topopah Tuff 8.5%

Second Character-Fuel Type

a= 1993 HEU
b= 1994 ATR
c= 1994 Shippingport
d= 1994 Graphite
e= 93.2 w% U235, 6.8% U238
f= Pure U235
g= Pure Pu239
h= 1993 INEEL LEU
I= 1993 INEEL Graphite
j= WG Plutonium (94w% Pu239 and 6w% Pu240)
k= Natural Uranium (99.2745% U238 and 0.72% U235)
l= U238 with 1% U235
m= U238 with 2% U235
n= U238 with 3% U235
o= U238 with 4% U235
p= U238 with 5% U235
x= 75w% U235 and 25w% Pu239
y= OKLO Composition (3.68w% U235 and 96.32w% Pu239)
z= WIPP Composition (87.7 w% U238, 4.75w% U235, and 7.55w% Pu239)

Third Character-Precipitate

a= Pure Uranium
b= Boltwoodite
c= Scheopite
d= Soddyite
f= Coffinite
h= Haiweeite
i= Pu (pure) + UO_2
j= Pu (pure) + Soddyite
k= Pu (pure) + Scheopite
l= PuO_2 + UO_2
m= PuO_2 (pure) + Soddyite
n= PuO_2 (pure) + Scheopite
o= Pure Plutonium
p= PuO_2
q= Weeksite
r= Ruthorfordine
t= Scheopite Dehydrate
u= Uraninite
w= Uranophane
x= UO_2 and other oxides in a fuel mixture
y= UO_2 and other elements in a fuel mixture

Fourth Character-Fissile Volume %

1= 0.01
2= 0.02
3= 0.03
4= 0.04
5= 0.05
6= 0.06
7= 0.07
8= 0.08
9= 0.09
a= 0.10
b= 0.20
c= 0.30
d= 0.40
e= 0.50
f= 0.60
g= 0.70
h= 0.80
i= 0.90
j= 1.00
k= 1.50
l= 2.00
m= 3.00
n= 4.00
o= 5.00
p= 6.00
q= 7.00
r= 8.00
s= 9.00
t= 10.00
u= 20.00
v= 30.00
w= 40.00
x= 50.00
y= 75.00
z= 100.00
#= other[1]

Fifth Character-Water Saturation

n= 0%
x= 1%
1= 10%
2= 20%
3= 30%
4= 40%
5= 50%
6= 60%
a= 65%
7= 70%
8= 80%
9= 90%
0= 100%
#= other

[1] The "other" character allows for specialization, even in the automated mode.

Sixth Character- Moderator Type

a= WIPP ERDA (Castile Brine)
g= WIPP G Seep Brine (Salado Brine)
h= Yucca Mountain UZ/High water
j= J-13 Well Water - Yucca Mountain
l= Yucca Mountain UZ/Low water
m= Yucca Mountain UZ/Mid water
n= None
p= Pure Water
s= WIPP Brine AIS (Culebra Brine)
u= Yucca Mountain UE-25p1

Seventh Character- Radius (cm)

a= 0.5 cm
b= 5 cm
c= 10 cm
d= 15 cm
e= 20 cm
f= 30 cm
g= 40 cm
h= 50 cm
i= 60 cm
j= 70 cm
k= 80 cm
l= 90 cm
m= 100 cm
n= 125 cm
o= 150 cm
p= 200 cm
q= 250 cm
r= 300 cm
s= 350 cm
t= 400 cm
u= 450 cm
v= 500 cm
w= 1000 cm
x= 2000 cm
y= 5000 cm
z= 10000 cm
%= other

Eighth Character-Reflector

n= None
a= Topopah Tuff (8.5%)
f= Topopah Tuff (13.9%)
h= Pure Water
b= WIPP salt
c= Pure Salt
d= Culebra
e= Bentonite
g= SiO_2
s= Sandstone
i= Graphite
q= Concrete (for half sphere)
m= None
o= Topopah Tuff (8.5%)
l= Topopah Tuff (13.9%)
k= Pure Water
w= WIPP salt
p= Pure Salt

```
r= Culebra
t= Bentonite
u= SiO2
v= Sandstone
j= Graphite
x= Concrete
```

A batch file is created at the same time as the input files to simplify running the study. After MCNP runs each file, the output is renamed to the input plus ".MOU". For example, output corresponding to poakajpa.MIN would be saved as poakajpa.MOU. This way, the input can be discarded and just the output kept. Also, with very little work, the input file can be recreated from the output file.

RKeff has a feature that allows it to strip out all pertinent information from the entire series of files. It will retrieve such things as k_{eff}, concentration, mass of fissile material, moderator to fissile atom ratio (MFR), and so on. All of these variables are placed in one table called MCNP_MAS.RES. This table can be sorted by another module of the RCS system and arranged for ease of plotting. If the parameters are plotted with k_{eff} as a function of concentration and radius or mass, this 3-D graph is known as a "criticality surface". The parameter study set up in MCNP_MAS.INP above would yield the surface shown Figures E.1-1.

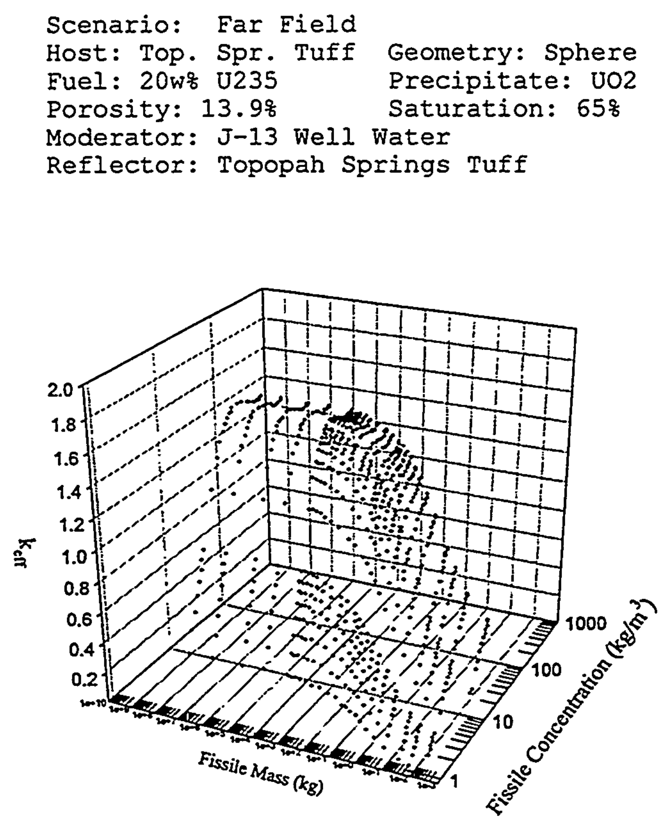


Figure E.1-1 Example criticality surface.

After MCNP_MAS.RES is created, RKeff uses a cubic interpolatory spline fit to calculate k_{eff} values. A buckling search is then used to identify the mass and radius values used to achieve criticality (k_{eff} =1.0). These buckling search values can then yield an "S-curve" (see Figure E.1-2). Multiple S-curves, for various enrichments in this case, can then be used to generate a "criticality saddle" (see Figure E.1-3). If the concentration, mass, or radius is less than the lowest point on this graph, the system will not be able to go critical. Further RKeff/MCNP output is listed in the remainder of this appendix, Section E.2 show an example benchmark of RKeff/MCNP and Section E.3 presents criticality results for CX models (see Tables 3.3-1 and 3.4-1).

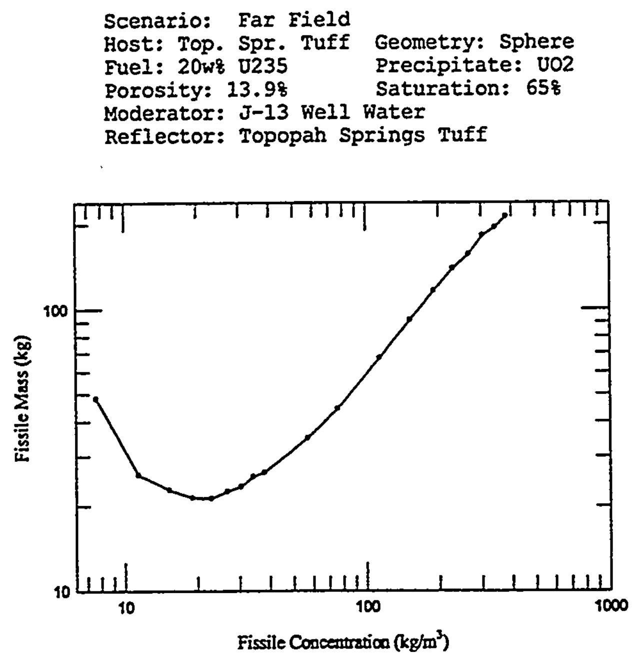


Figure E.1-2 Example criticality S-curve (far-field case).

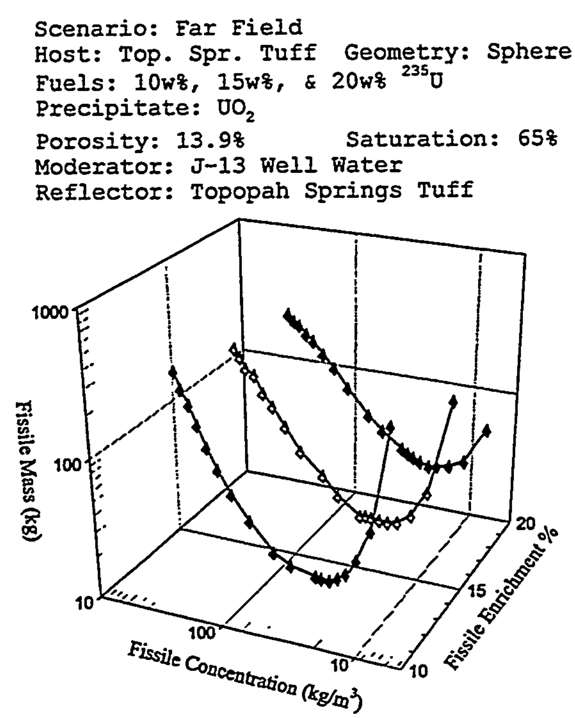


Figure E.1-3 Example criticality saddle for three different fuel enrichments (far-field case).

E.2 Example Problem (Generation of ^{239}Pu S-Curve)

As a quality check on the computational process used to generate criticality S-curves, a special set of criticality calculations were performed for a system comprised of ^{239}Pu and pure water. This test case was chosen since data already exists in the open literature. Figure E.2-1 presents the results obtained with the RKeff/MCNP™ system compared with existing published data. From this figure it can be noted that the computational analysis (k_{eff} calculations and the use of cubic interpolating splines to determine criticality buckling) was performed adequately. Inaccuracies are due to the limited number of "histories" used in the MCNP eigenvalue calculations. To expedite the computational work and allow a large set of static criticality models to be investigated, the Monte Carlo calculations were performed for 100,000 histories. From Figure E.2-2 it can be seen that diminishing returns are obtained for calculations with greater than 100,000 histories due to the relationship of the standard deviation with histories (e.g., σ follows a Poisson distribution where σ is inversely proportional to the square root of the number of histories).

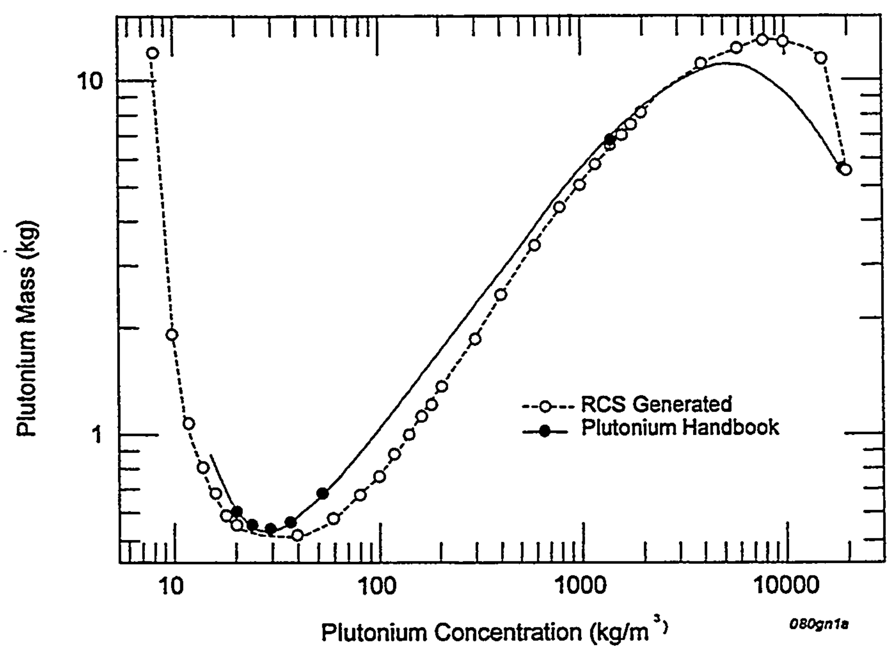


Figure E.2-1 Criticality S-curve computed using RKeff and MCNP (computed values for critical fissile mass and concentrations compared to values from Clayton 1980).

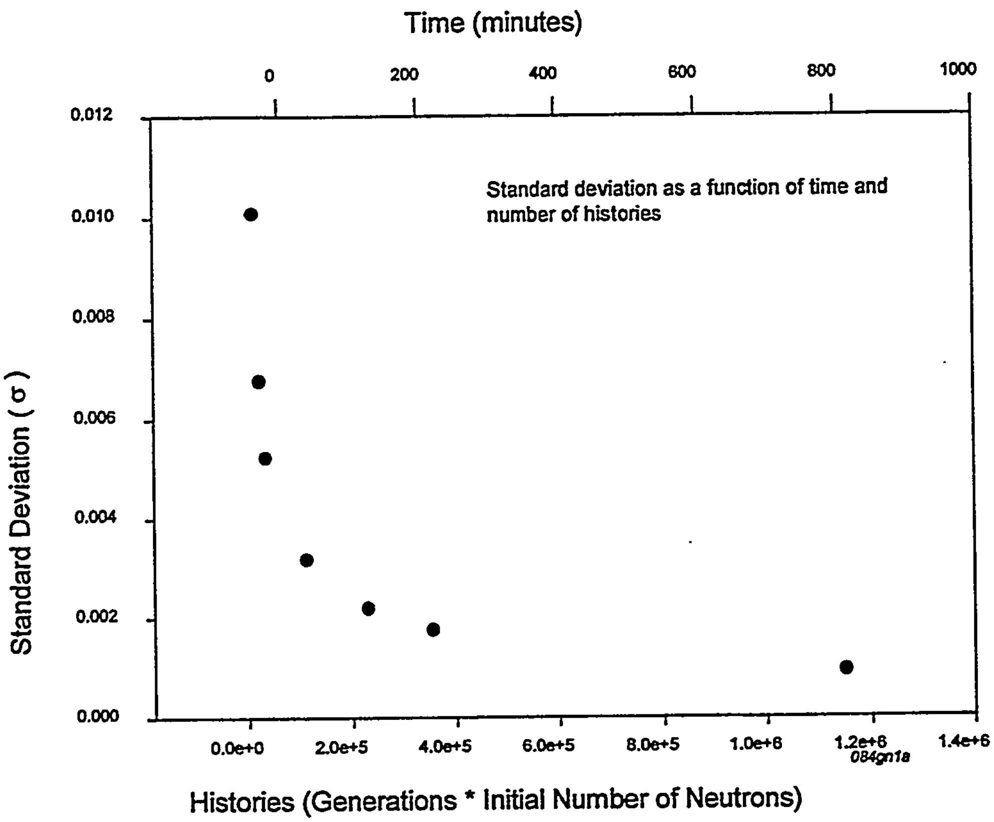


Figure E.2-2 Standard deviation as a function of number of histories for typical far-field criticality calculations (computations performed with MCNP are performed at 100,000 histories).

E.3 Graphical Results of CX Model Calculations

CX-01 (see Tables 3.3-1 and 3.4-1 for list of CX models)

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 5wt% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

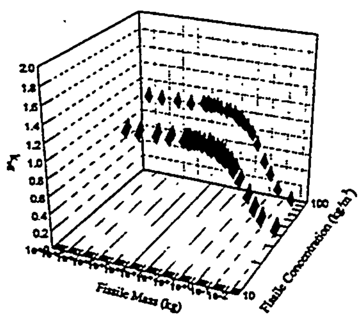


Figure E.3-1 Criticality surface for CX-01 model

[Not enough points for S-curve]

CX-02

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 10wt% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

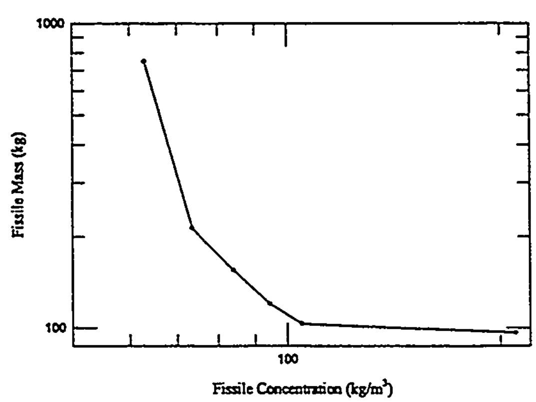


Figure E.3-2 Criticality S-curve for CX-02 model

CX-03

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 15w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

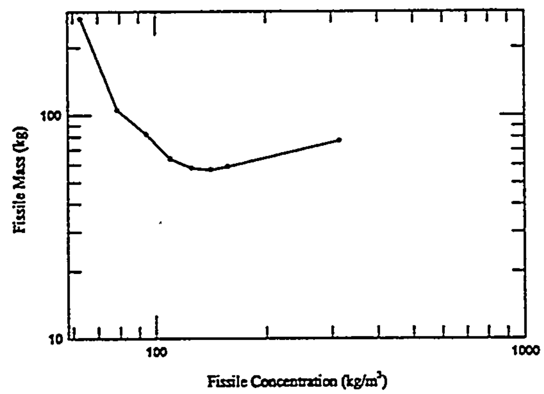


Figure E.3-3 Criticality S-curve for CX-03 model

CX-04

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 20w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

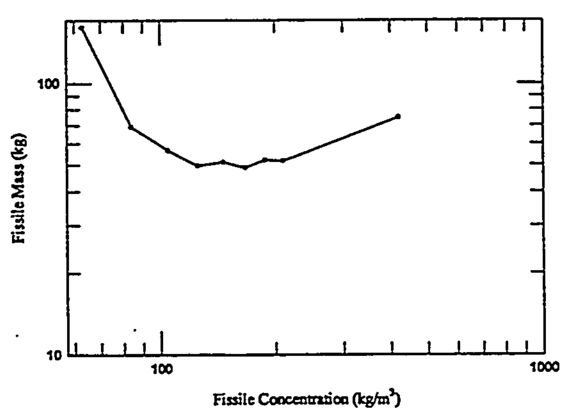


Figure E.3-4 Criticality S-curve for CX-04 model

CX-05

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 25w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

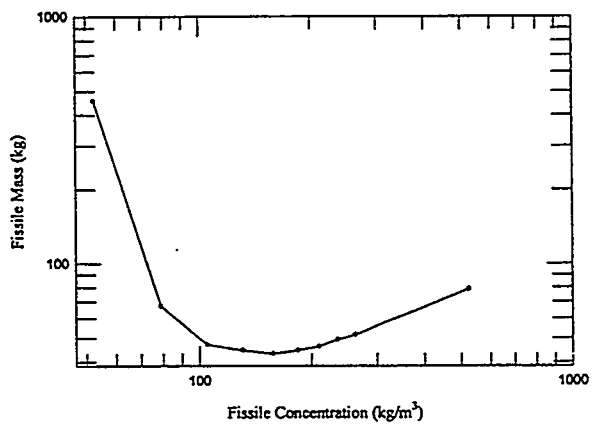


Figure E.3-5 Criticality S-curve for CX-05 model

CX-06

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 5w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

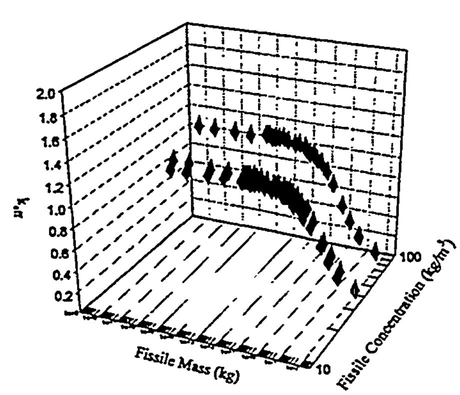


Figure E.3-6 Criticality surface for CX-05 model

[Not enough points for S-curve]

CX-07

```
Scenario:  Near Field
Host: Rust          Geometry: Hemisphere
Fuel: 10w% 235U     Precipitate: UO2
Porosity: 20%       Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

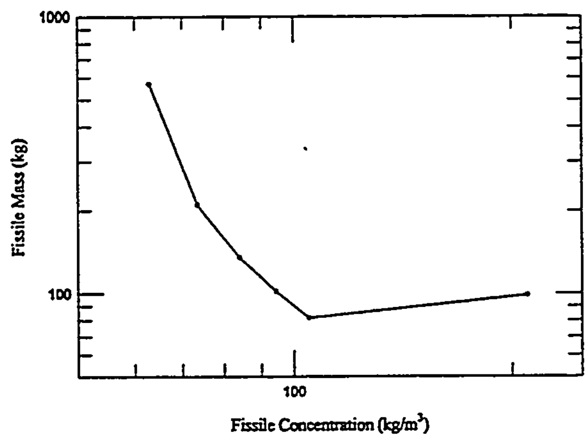


Figure E.3-7 Criticality S-curve for CX-07 model

CX-08

```
Scenario:  Near Field
Host: Rust          Geometry: Hemisphere
Fuel: 15w% 235U     Precipitate: UO2
Porosity: 20%       Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

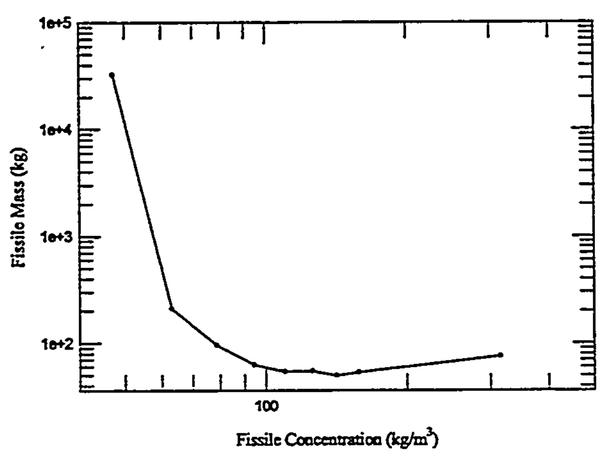


Figure E.3-8 Criticality S-curve for CX-08 model

CX-09

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 20w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

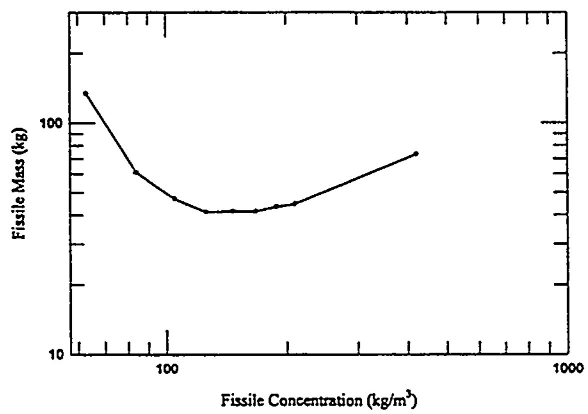


Figure E.3-9 Criticality S-curve for CX-09 model

CX-10

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 25w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

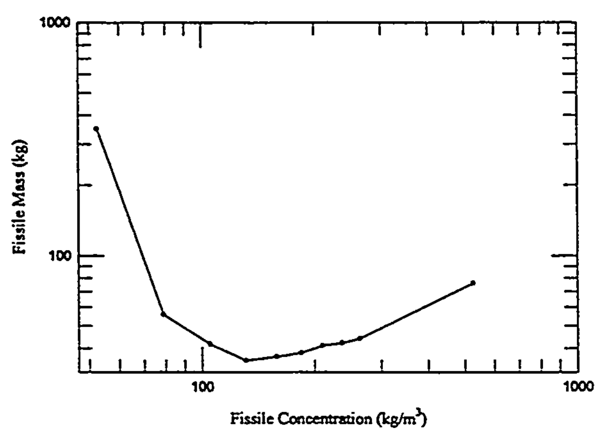


Figure E.3-10 Criticality S-curve for CX-10 model

CX-11

```
Scenario: Near Field
Host: Rust          Geometry: Hemisphere
Fuel: 5wt% 235U     Precipitate: UO2
Porosity: 20%       Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

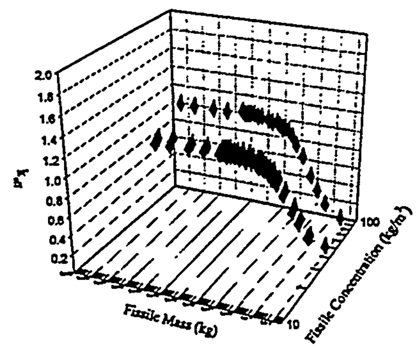


Figure E.3-11 Criticality surface for CX-11 model

[Not enough points for S-curve]

CX-12

```
Scenario:  Near Field
Host: Rust           Geometry: Hemisphere
Fuel: 10w% 235U     Precipitate: UO2
Porosity: 20%       Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

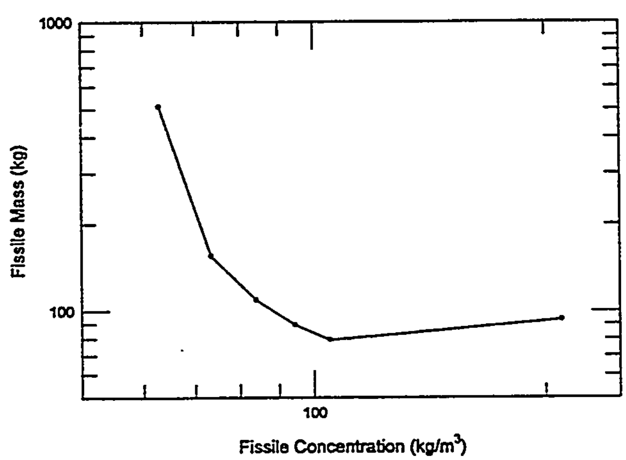


Figure E.3-12 Criticality S-curve for CX-12 model

CX-13

```
Scenario:  Near Field
Host: Rust             Geometry: Hemisphere
Fuel: 15w% 235U      Precipitate: UO2
Porosity: 20%          Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

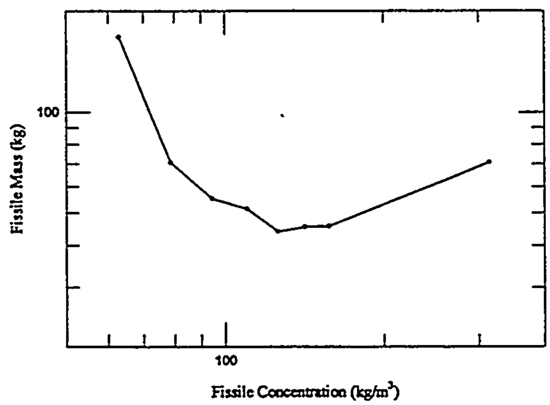


Figure E.3-13 Criticality S-curve for CX-13 model

CX-14

```
Scenario:  Near Field
Host: Rust             Geometry: Hemisphere
Fuel: 20w% 235U      Precipitate: UO2
Porosity: 20%          Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

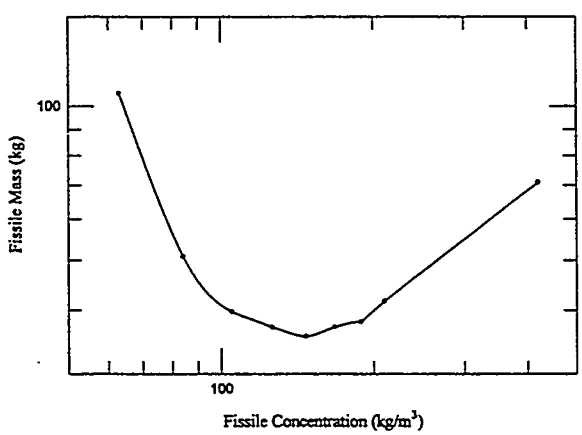


Figure E.3-14 Criticality S-curve for CX-14 model

CX-15

Scenario: Near Field
Host: Rust　　Geometry: Hemisphere
Fuel: 25w% ^{235}U　　Precipitate: UO_2
Porosity: 20%　　Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

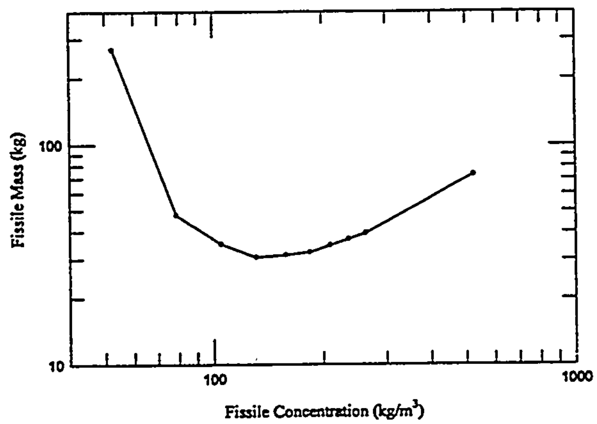


Figure E.3-15 Criticality S-curve for CX-15 model

CX-16

Scenario: Near Field
Host: Rust　　Geometry: Hemisphere
Fuel: 5w% ^{235}U　　Precipitate: UO_2
Porosity: 20%　　Saturation: 80%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

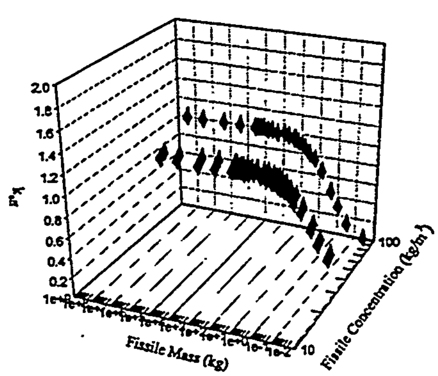


Figure E.3-16 Criticality surface for CX-16 model

[Not enough points for S-curve]

CX-17

```
Scenario: Near Field
Host: Rust              Geometry: Hemisphere
Fuel: 10w% 235U        Precipitate: UO2
Porosity: 20%          Saturation: 80%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

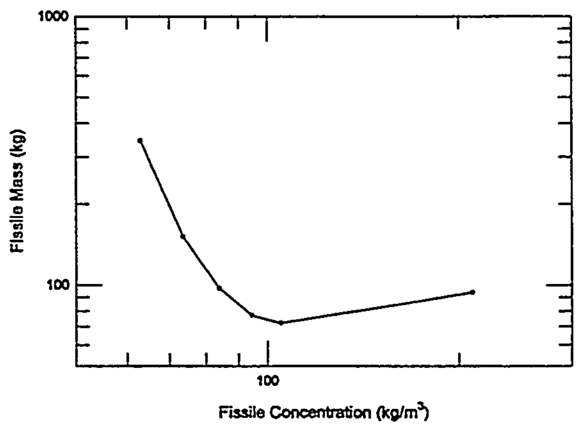


Figure E.3-17 Criticality S-curve for CX-17 model

CX-18

```
Scenario:  Near Field
Host: Rust              Geometry: Hemisphere
Fuel: 15w% 235U        Precipitate: UO2
Porosity: 20%          Saturation: 80%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

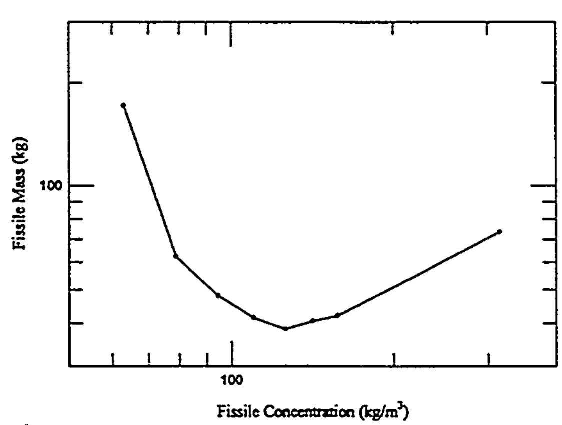


Figure E.3-18 Criticality S-curve for CX-18 model

CX-19

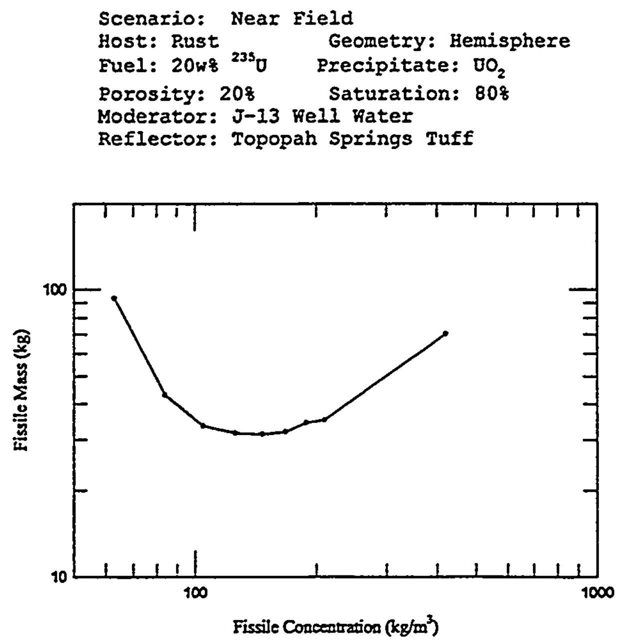


Figure E.3-19 Criticality S-curve for CX-19 model

CX-20

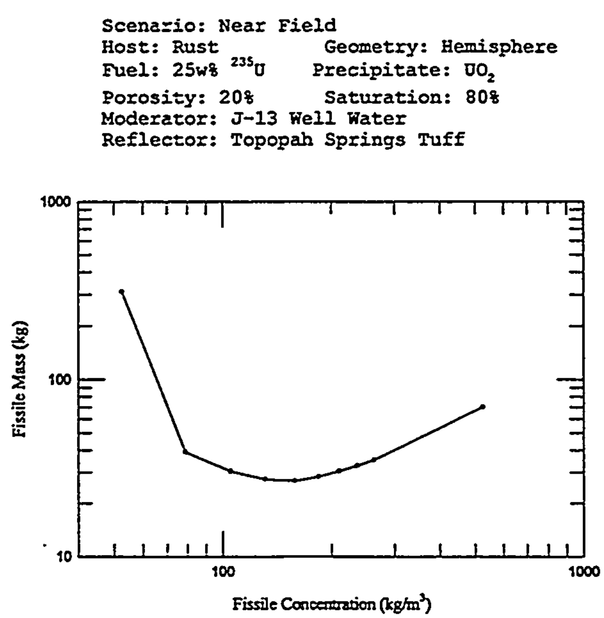


Figure E.3-20 Criticality S-curve for CX-20 model

CX-21

[Not enough points for S-curve]

CX-22

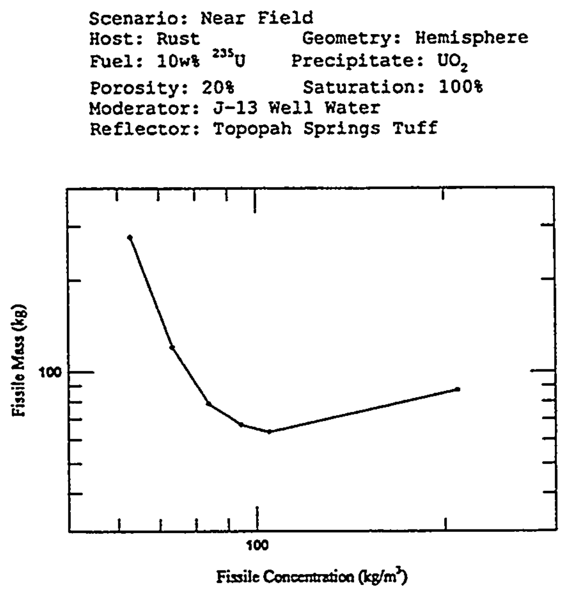


Figure E.3-22 Criticality S-curve for CX-22 model

CX-23

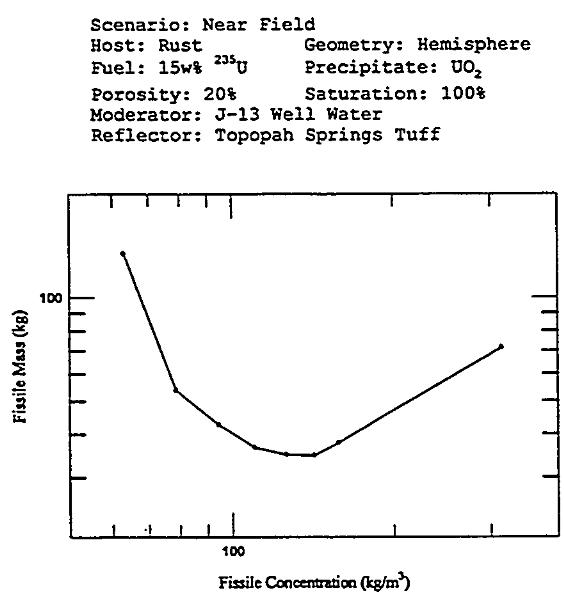


Figure E.3-23 Criticality S-curve for CX-23 model

CX-24

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 20w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 100%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

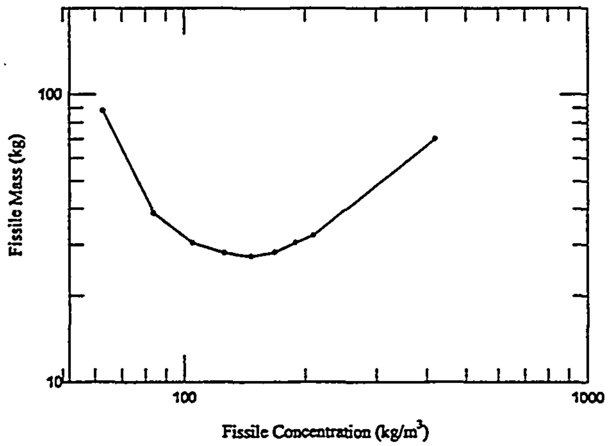


Figure E.3-24 Criticality S-curve for CX-24 model

CX-25

Scenario: Near Field
Host: Rust Geometry: Hemisphere
Fuel: 25w% ^{235}U Precipitate: UO_2
Porosity: 20% Saturation: 100%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

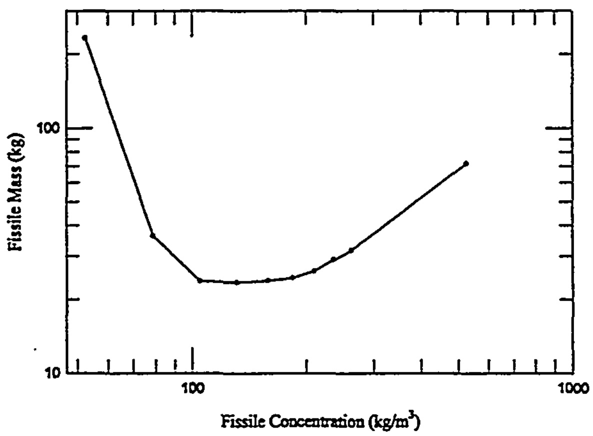


Figure E.3-25 Criticality S-curve for CX-25 model

CX-26

```
Scenario: Near Field
Host: Rust            Geometry: Hemisphere
Fuel: HEU             Precipitate: UO2
Porosity: 20%         Saturation: 100%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

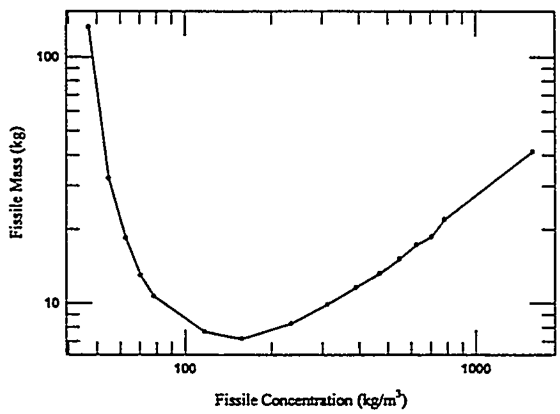


Figure E.3-26 Criticality S-curve for CX-26 model

CX-27

[No S-curve available]

CX-28

```
Scenario:  Far Field
Host: TST             Geometry: Sphere
Fuel: LEU             Precipitate: UO2
Porosity: 13.9%    Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

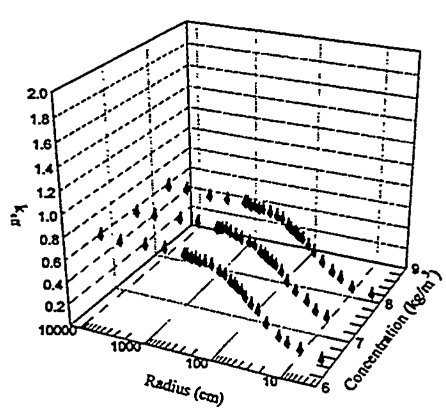


Figure E.3-28 Criticality surface for CX-28 model

CX-29

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: HEU Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

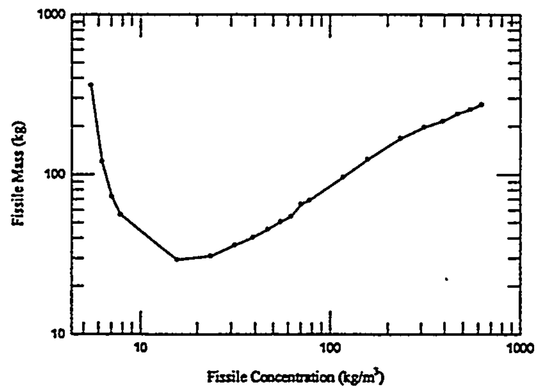


Figure E.3-29 Criticality S-curve for CX-29 model

CX-30

[Does not reach criticality]

CX-31

[Does not reach criticality]

CX-32

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 3wt% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

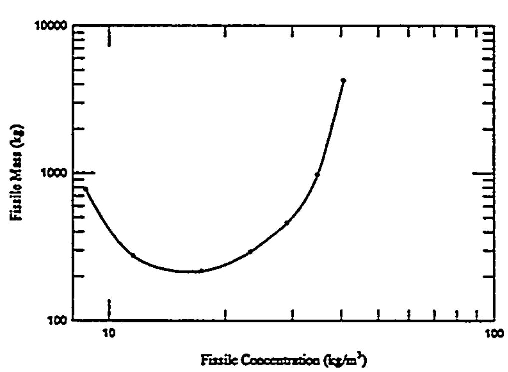


Figure E.3-32 Criticality S-curve for CX-32 model

CX-33

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 4w% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

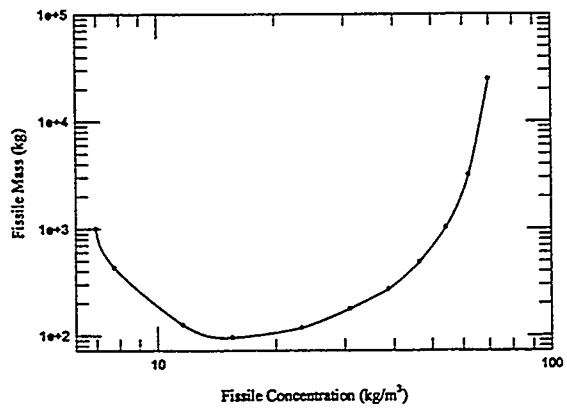


Figure E.3-33 Criticality S-curve for CX-33 model

CX-34

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 5w% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

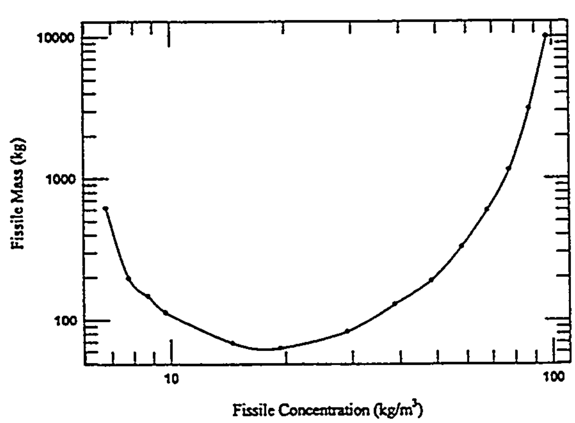


Figure E.3-34 Criticality S-curve for CX-34 model

CX-35

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 10w% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

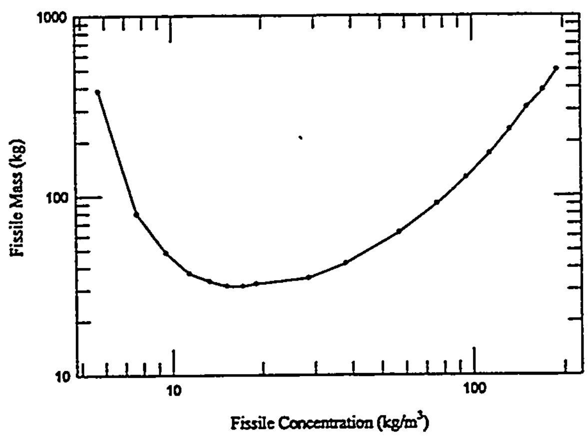


Figure E.3-35 Criticality S-curve for CX-35 model

CX-36

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 15w% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

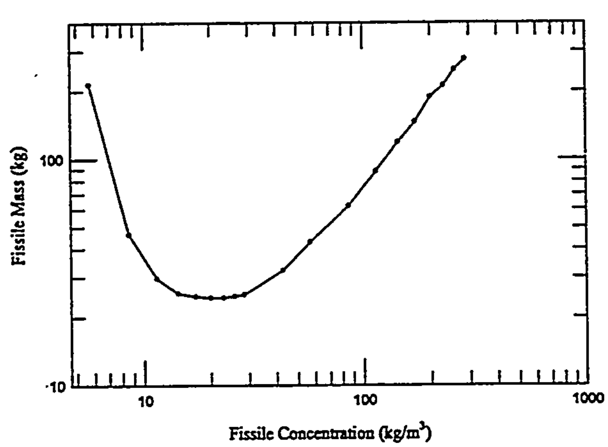


Figure E.3-36 Criticality S-curve for CX-36 model

CX-37

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 20w% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

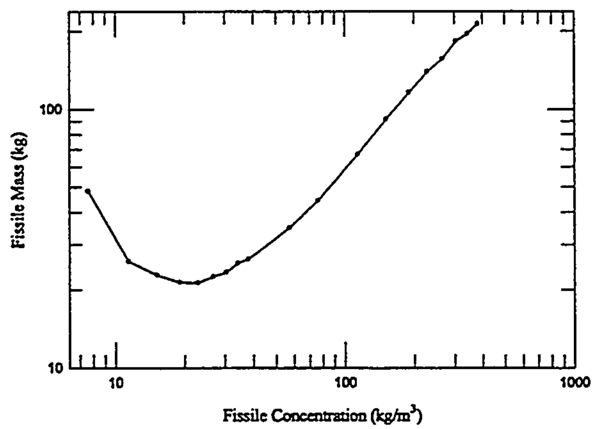


Figure E.3-37 Criticality S-curve for CX-37 model

CX-38

Scenario: Far Field
Host: Top. Spr. Tuff Geometry: Sphere
Fuel: 25w% ^{235}U Precipitate: UO_2
Porosity: 13.9% Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

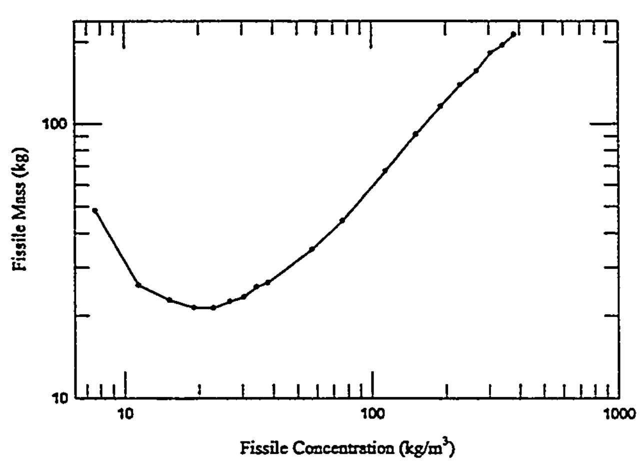


Figure E.3-38 Criticality S-curve for CX-38 model

CX-39

```
Scenario:  Far Field
Host: Top. Spr. Tuff  Geometry: Sphere
Fuel: HEU             Precipitate: UO2
Porosity: 13.9%       Saturation: 65%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

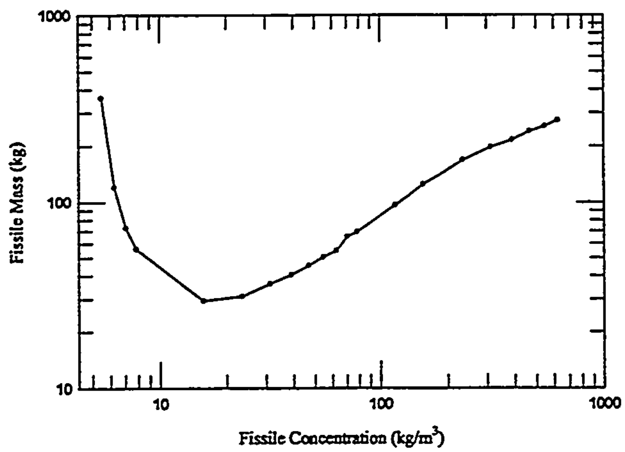


Figure E.3-39 Criticality S-curve for CX-39 model

CX-40

[No S-curve available]

CX-41

```
Scenario: Near Field
Host: Concrete      Geometry: Hemisphere
Fuel: 5w% 235U      Precipitate: UO2
Porosity: 10%       Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff
```

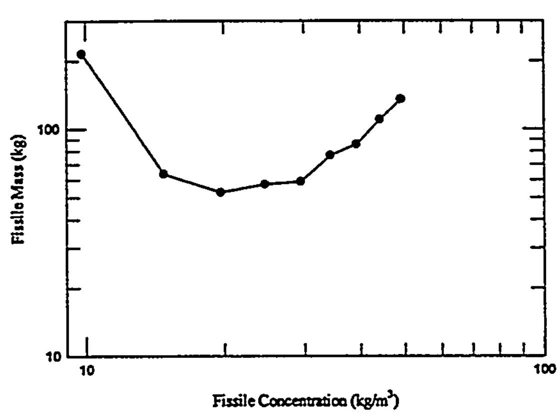


Figure E.3-41 Criticality S-curve for CX- 41 model

CX-42

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 10w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

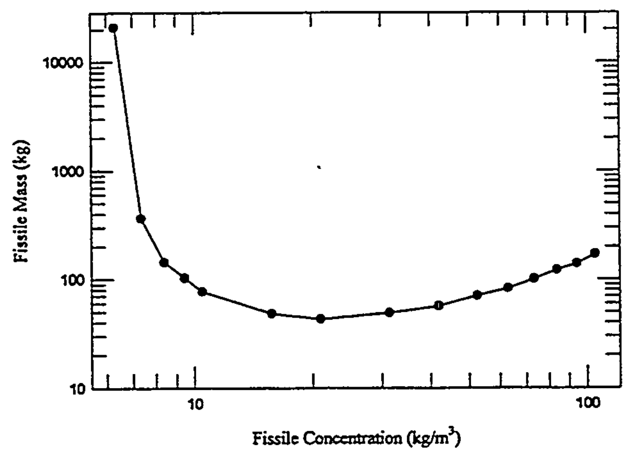


Figure E.3-42 Criticality S-curve for CX-42 model

CX-43

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 15w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

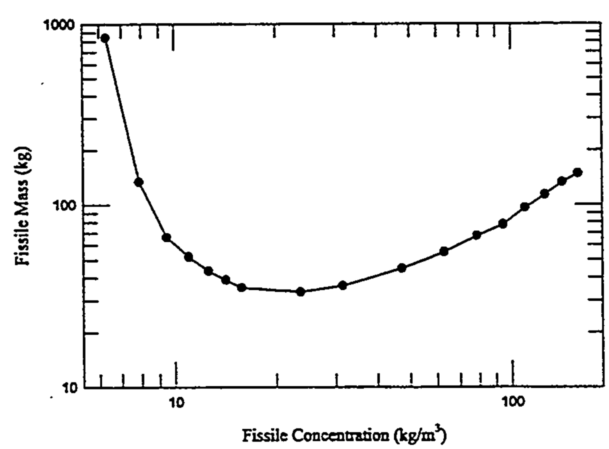


Figure E.3-43 Criticality S-curve for CX-43 model

CX-44

Scenario: Near Field
Host: Concrete　　Geometry: Hemisphere
Fuel: 20w% ^{235}U　　Precipitate: UO_2
Porosity: 10%　　Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

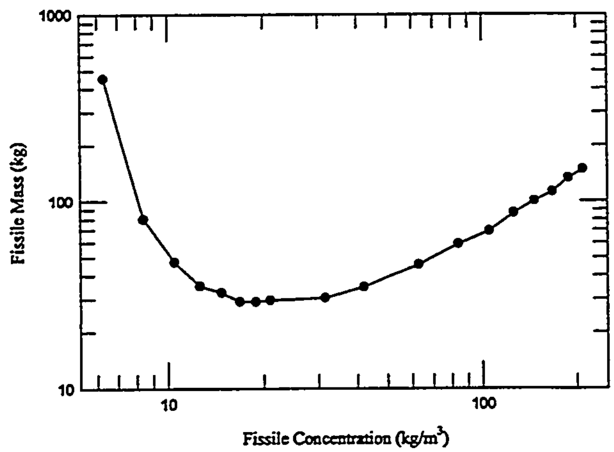


Figure E.3-44 Criticality S-curve for CX-44 model

CX-45

Scenario: Near Field
Host: Concrete　　Geometry: Hemisphere
Fuel: 25w% ^{235}U　　Precipitate: UO_2
Porosity: 10%　　Saturation: 20%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

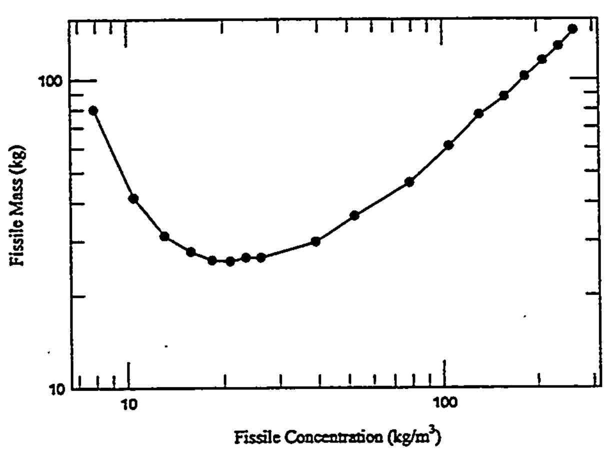


Figure E.3-45 Criticality S-curve for CX-45 model

CX-46

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 5w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

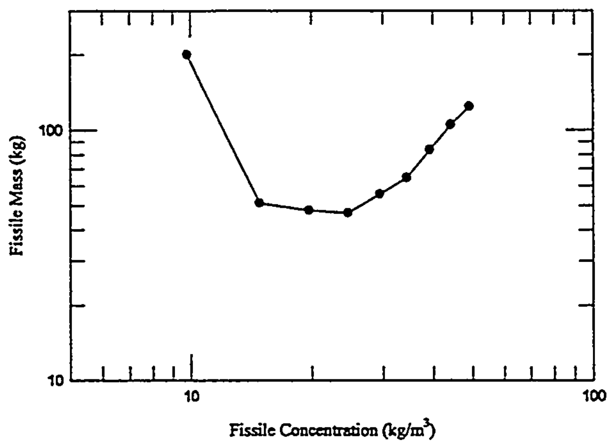


Figure E.3-46 Criticality S-curve for CX-46 model

CX-47

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 10w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

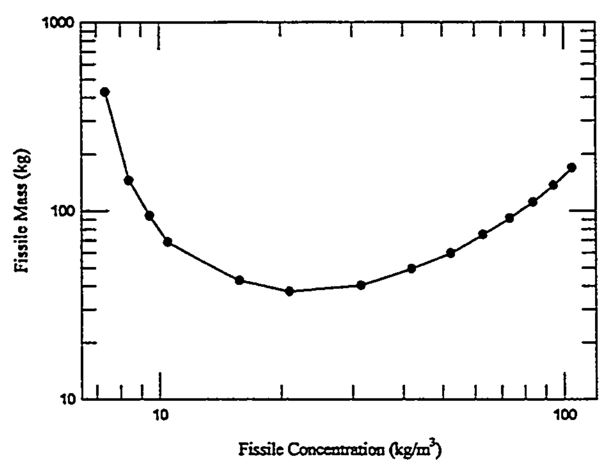


Figure E.3-47 Criticality S-curve for CX-47 model

CX-48

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 15w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

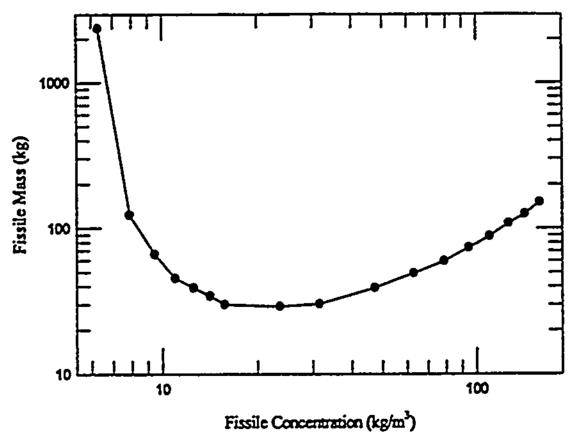


Figure E.3-48 Criticality S-curve for CX-48 model

CX-49

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 20w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

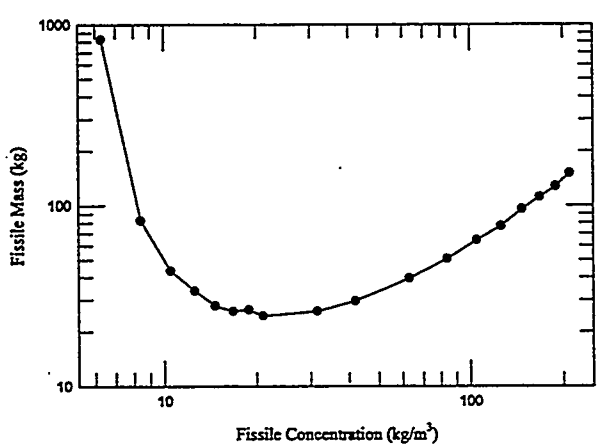


Figure E.3-49 Criticality S-curve for CX-49 model

CX-50

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 25 w% ^{235}U Precipitate: UO2
Porosity: 10%   Saturation: 40%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

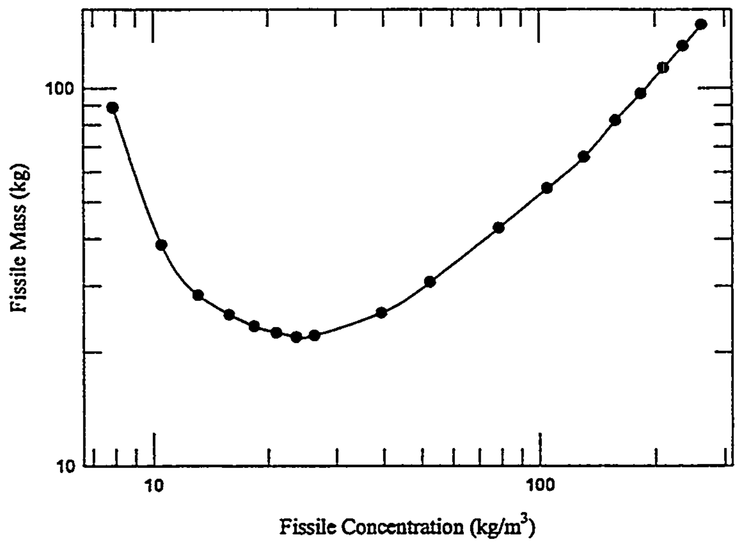


Figure E.3-50 Criticality S-curve for CX-50 model

CX-51

Scenario: Near Field
Host: Concrete   Geomtery: Hemisphere
Fuel: 5w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

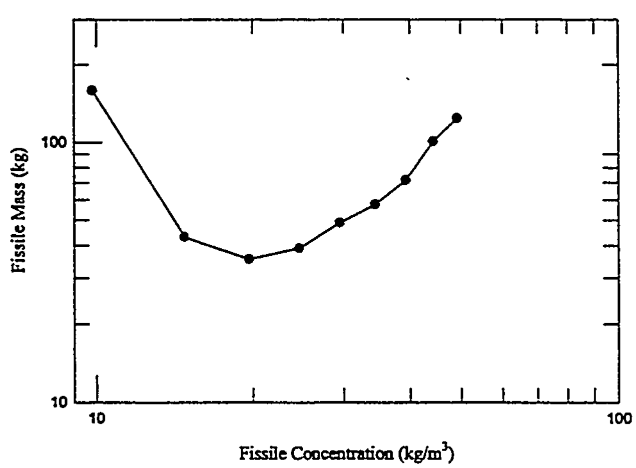


Figure E.3-51 Criticality S-curve for CX-51 model

CX-52

Scenario: Near Field
Host: Concrete — Geometry: Hemisphere
Fuel: 10w% ^{235}U — Precipitate: UO_2
Porosity: 10% — Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

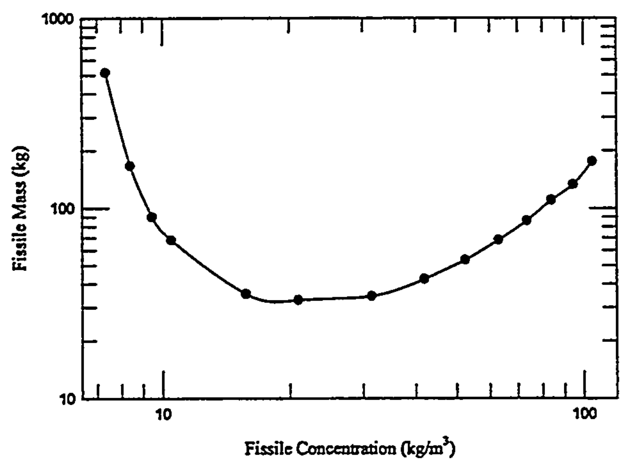


Figure E.3-52 Criticality S-curve for CX-52 model

CX-53

Scenario: Near Field
Host: Concrete — Geometry: Hemisphere
Fuel: 15w% ^{235}U — Precipitate: UO_2
Porosity: 10% — Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

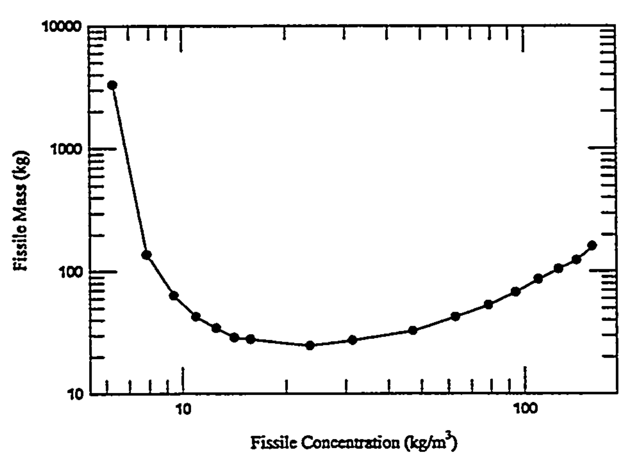


Figure E.3-53 Criticality S-curve for CX-53 model

CX-54

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 20w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

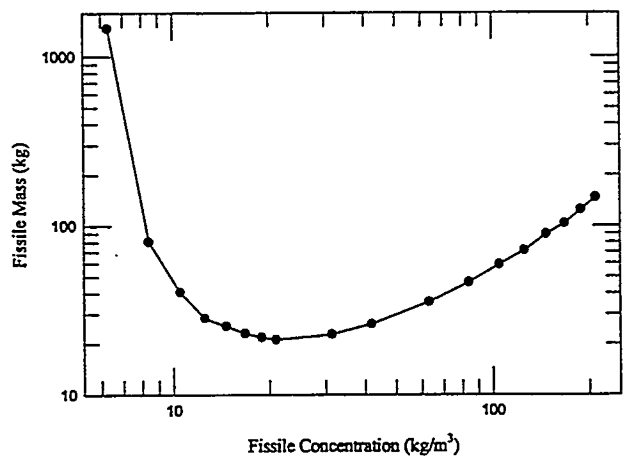


Figure E.3-54 Criticality S-curve for CX-54 model

CX-55

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel:25w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 60%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

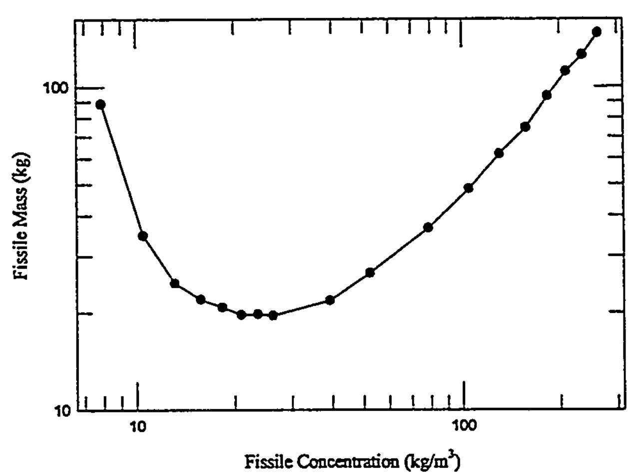


Figure E.3-55 Criticality S-curve for CX-55 model

CX-56

Scenario: Near Field
Host: Concrete Geometry: Hemisphere
Fuel: 5w% ^{235}U Precipitate: UO_2
Porosity: 10% Saturation: 80%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

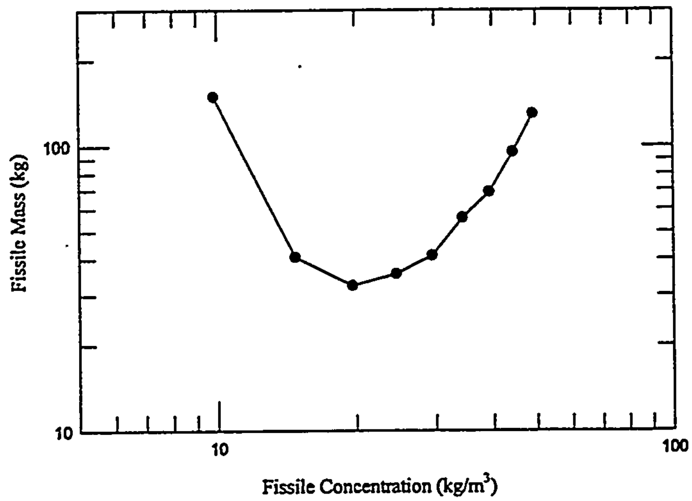


Figure E.3-56 Criticality S-curve for CX-56 model

CX-57

Scenario: Near Field
Host: Concrete Geometry: Hemisphere
Fuel: 10 w% ^{235}U Precipitate: UO_2
Porosity: 10% Saturation: 80%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

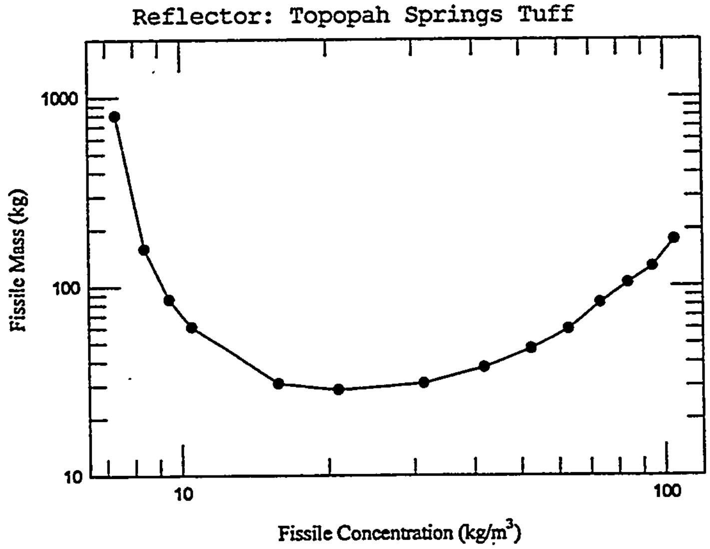


Figure E.3-57 Criticality S-curve for CX-57 model

CX-58

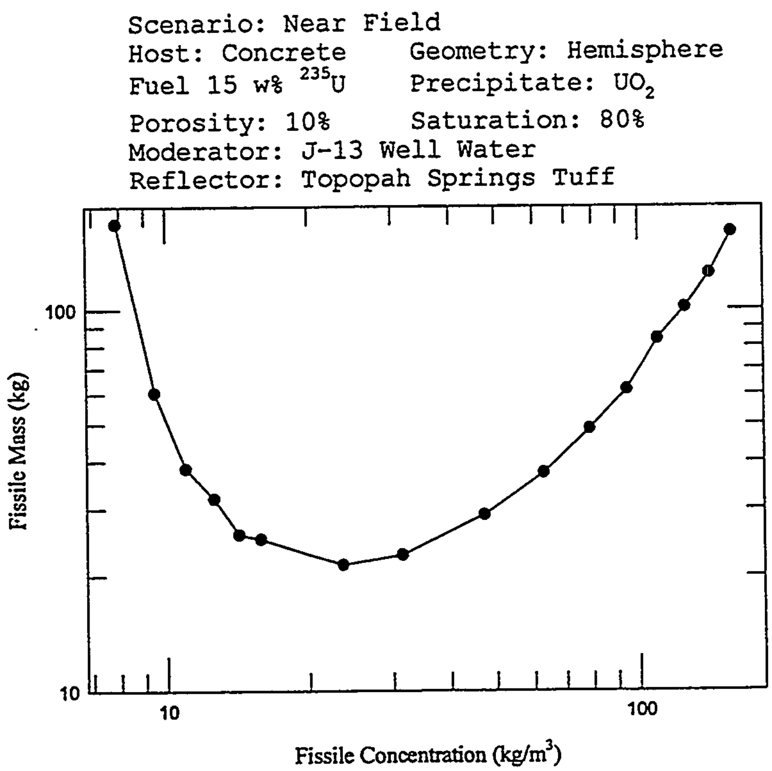


Figure E.3-58 Criticality S-curve for CX-58 model

CX-59

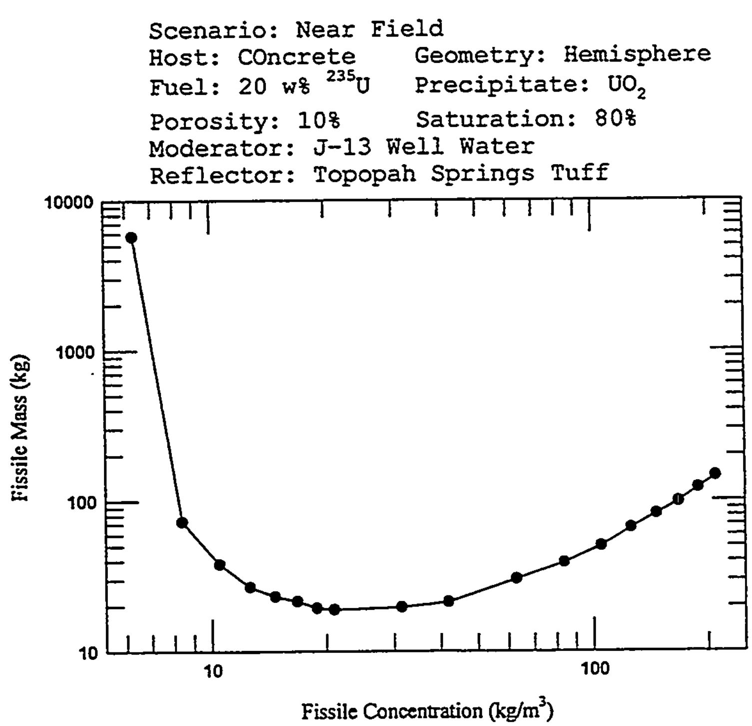


Figure E.3-59 Criticality S-curve for CX-59 model

CX-60

Scenario: Near Field
Host: Concrete   Geometry: Hemisphere
Fuel: 25 w% ^{235}U Precipitate: UO_2
Porosity: 10%   Saturation: 80%
Moderator: J-13 Well Water
Reflector: Topopah Springs Tuff

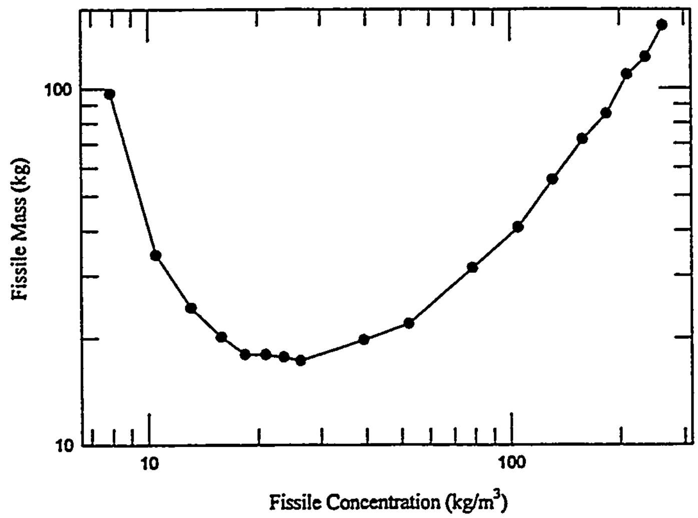


Figure E.3-60 Criticality S-curve for CX-60 model

CX-X1

Benchmark
Host: Water   Geometry: Sphere
Fuel: Pure ^{235}U Precipitate: U
Porosity: 100%   Saturation: 100%
Moderator: H_2O Reflector: H_2O

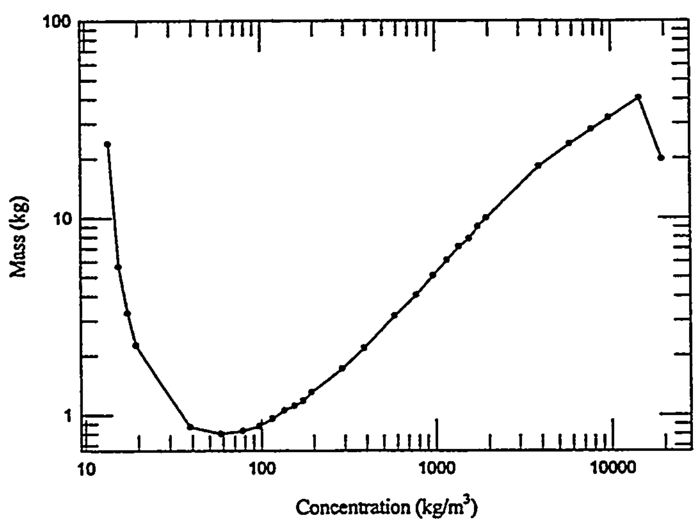


Figure E.3-X1 Benchmark criticality S-curve for uranium, with reflector

CX-X2

Benchmark
Host: Water — Geometry: Sphere
Fuel: Pure ^{235}U — Precipitate: U
Porosity: 100% — Saturation: 100%
Moderator: H_2O
Reflector: None

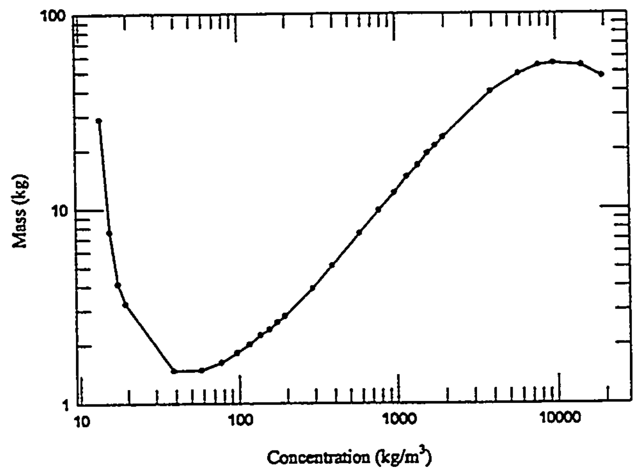


Figure E.3-X2 Benchmark criticality S-curve for uranium without reflector

CX-X3

Benchmark
Host: Pure Water — Geometry: Sphere
Fuel: Pure ^{239}Pu — Precipitate: ^{239}Pu
Saturation: 100%
Moderator: Pure Water
Reflector Pure Water

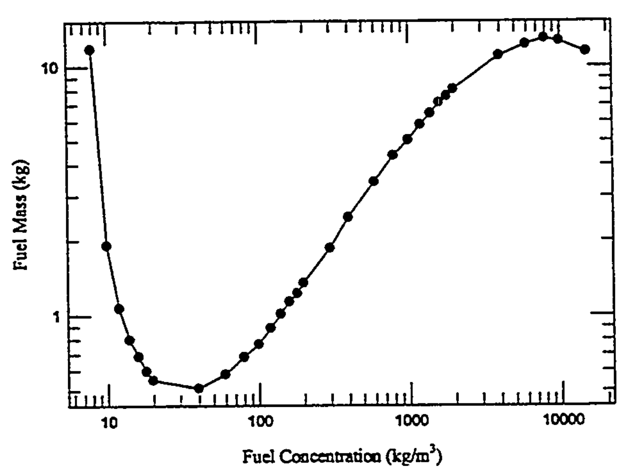


Figure E.3-X3 Benchmark criticality S-curve for plutonium

Table E.3-1. Key Results Shown in Figures E.3-1 Through E.3-60

CX number	Host Material	Fuel Enrichment (weight % ^{235}U)	Saturation
01	rust	5	20
02	rust	10	20
03	rust	15	20
04	rust	20	20
05	rust	25	20
06	rust	5	40
07	rust	10	40
08	rust	15	40
09	rust	20	40
10	rust	25	40
11	rust	5	60
12	rust	10	60
13	rust	15	60
14	rust	20	60
15	rust	25	60
16	rust	5	80
17	rust	10	80
18	rust	15	80
19	rust	20	80
20	rust	25	80
21	rust	5	100
22	rust	10	100
23	rust	15	100
24	rust	20	100
25	rust	25	100
26	rust	HEU[1]	100
27	concrete	HEU	100
28	TST[2]	LEU[3]	65
29	TST	HEU	65
30	TST	1	65
31	TST	2	65
32	TST	3	65
33	TST	4	65
34	TST	5	65
35	TST	10	65

[1] Highly enriched uranium (80 w% ^{235}U)
[2] Topopah Springs Tuff
[3] Low enriched uranium (0.8 w% ^{235}U)

36	TST	15	65
37	TST	20	65
38	TST	25	65
39	**SAME**	**AS**	**#29**
40	TST	100	65
41	Concrete	5	20
42	Concrete	10	20
43	Concrete	15	20
44	Concrete	20	20
45	Concrete	25	20
46	Concrete	5	40
47	Concrete	10	40
48	Concrete	15	40
49	Concrete	20	40
50	Concrete	25	40
51	Concrete	5	60
52	Concrete	10	60
53	Concrete	15	60
54	Concrete	20	60
55	Concrete	25	60
56	Concrete	5	80
57	Concrete	10	80
58	Concrete	15	80
59	Concrete	20	80
60	Concrete	25	80
Benchmarks			
X1	Pure Water (water reflector)	Pure ^{235}U	100
X2	Pure Water (no reflector)	Pure ^{235}U	100
X3	Pure Water (water reflector)	Pure ^{239}Pu	100

APPENDIX F

Uncoupled Nuclear Dynamics (UDX) Model

F.1 Large Reactivity Excursion Compared to Nordheim-Fuchs Approximation

This simple uncoupled nuclear dynamics calculation was designed to test the nuclear dynamics model (using the NARK code) with the Nordheim-Fuchs (NF) approximation for a high reactivity nuclear excursion. The computational results from NARK demonstrate that the numerical model correctly simulates rapid self-shutdown of a fast system with significant reactivity temperature feedback effects. Using an adiabatic model, the NF approximation can be used to predict the peak power response resulting from a super-prompt critical condition (reactivity $\rho_0 > \beta$, where β is the delayed neutron fraction). The essence of the Nordheim-Fuchs approximation is that self-limiting power excursions occur very rapidly, and thus the delayed neutron production and extraneous source neutrons may be neglected entirely (Hetrick, 1971: pg. 164). According to the NF approximation, the peak power response may be written as follows:

$$N = \frac{l \cdot \omega^2 \cdot M \cdot c_P}{2 \cdot |\alpha_D|} \tag{F.1-1}$$

$$\omega = \frac{\rho_0 - \beta}{l} \tag{F.1-2}$$

The NARK benchmark NF problems were limited to a thermal and fast reactor assembly using nuclear data for ^{235}U as the fissile material. Equilibrium initial conditions were applied to the point kinetics equations, as well as the initial condition of $T(0) = 0$.

The thermal assembly considered the following ratios, $\frac{l}{\beta} = 10^{-1}, \frac{|\alpha_D|}{Mc_P\beta} = 1$, and $N(0) = 10^{-6}$ for three initial reactivity bursts equivalent to ω= 8, 10, and 12 sec^{-1} (corresponding to super-prompt critical reactivity insertions of \$1.756, \$1.964, and \$2.170) (Hetrick, 1971: pg. 172). Using the NF approximation, peak power values of 3.20, 5.00, and 7.20 are predicted. The NARK benchmark NF calculations for the ^{235}U thermal reactor assembly displaying power versus time is shown in Figure F.1-1. As can be seen from this figure, the NARK numerical solution agrees with the NF solution. The fast reactor assembly considered the following ratios, $\frac{l}{\beta} = 10^{-6}, \frac{|\alpha_D|}{Mc_P\beta} = 1$, and $N(0) = 10^{-6}$ for three initial reactivity bursts equivalent to ω = 2400, 3000, and 3600 sec^{-1} (which are slightly above prompt critical reactivity insertions of \$1.00223, \$1.00286, and \$1.00349) (Hetrick, 1971: pg. 173). The NF approximation predicts fast assembly peak power values of 2.88, 4.50, and 6.50. The NARK benchmark NF calculations for the ^{235}U fast reactor assembly displaying power versus time is shown in Figure F.1-2. Again, the NARK numerical solution is in good agreement with solution obtained with the NF approximation.

It should be noted that the Nordheim-Fuchs solution, Equations F.1-1 and F.1-2, is an approximation to nuclear dynamics using an adiabatic heat-loss model. Therefore, the NARK model is not expected to exactly match the NF predictions because the NARK model is both physically and mathematically more rigorous. Also the NARK results shown in Figures F.1-1 and F.1-2 are similar to published figures (Hetrick, 1971: pg. 173-174).

F.2 Small Reactivity Excursions using the Nuclear Dynamics Model

To further test the NARK nuclear dynamics model in the uncoupled mode, a small reactivity excursion (an initial reactivity of $\rho_0 < \beta$) was used to demonstrate appropriate simulation of reactivity temperature feedback effects for slow nuclear excursions. Again, using the adiabatic heat loss model, the code was run simulating two slow power excursions having similar initial reactivities. This small reactivity excursion study was limited to a thermal and fast assembly using nuclear data for ^{235}U as the fissile material. Equilibrium initial conditions were applied to the nuclear kinetics equations, as well as the initial condition of $\dot{T}(0) = 0$. The thermal assembly considered the following ratios:, $\frac{l}{\beta} = 10^{-1}, \frac{|\alpha_D|}{Mc_p\beta} = 1$ and $N(0) = 10^{-6}$ for the small initial reactivity excursion of 0.4029 (in dollars worth of reactivity, *i.e.* $\$\rho = \frac{\rho_0}{\beta} = 0.4029$). The fast assembly considered the following ratios: $\frac{l}{\beta} = 10^{-6}, \frac{|\alpha_D|}{Mc_p\beta} = 1$, and $N(0) = 10^{-6}$ for the small initial reactivity excursion of 0.3992 (in dollars worth of reactivity, *i.e.* $\$\rho = \frac{\rho_0}{\beta} = 0.3992$). Shown in Figures F.2-1 and F.2-2 are the power histories and the power versus reactivity responses for the overlayed thermal and fast assemblies of ^{235}U. The NARK results shown in Figure F.2-2 are important because they depict the reactivity dependence on power and are in exact synchronization with the rise and fall of the corresponding power excursions shown in Figure F.2-1.

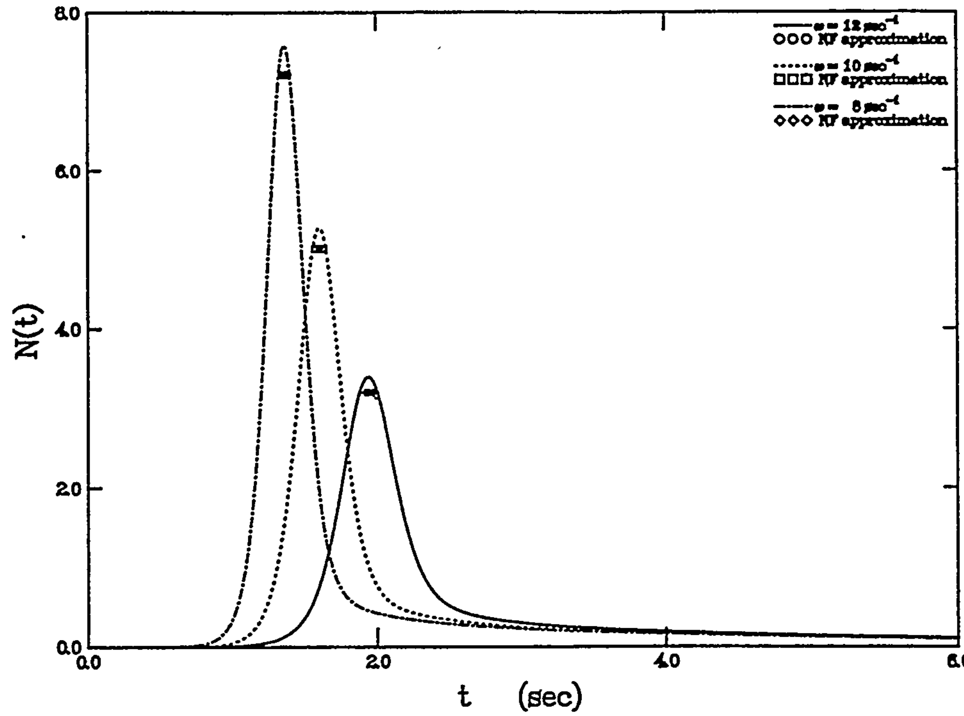


Figure F.1-1. Power histories (watts) for ^{235}U thermal assembly and NF approximation.

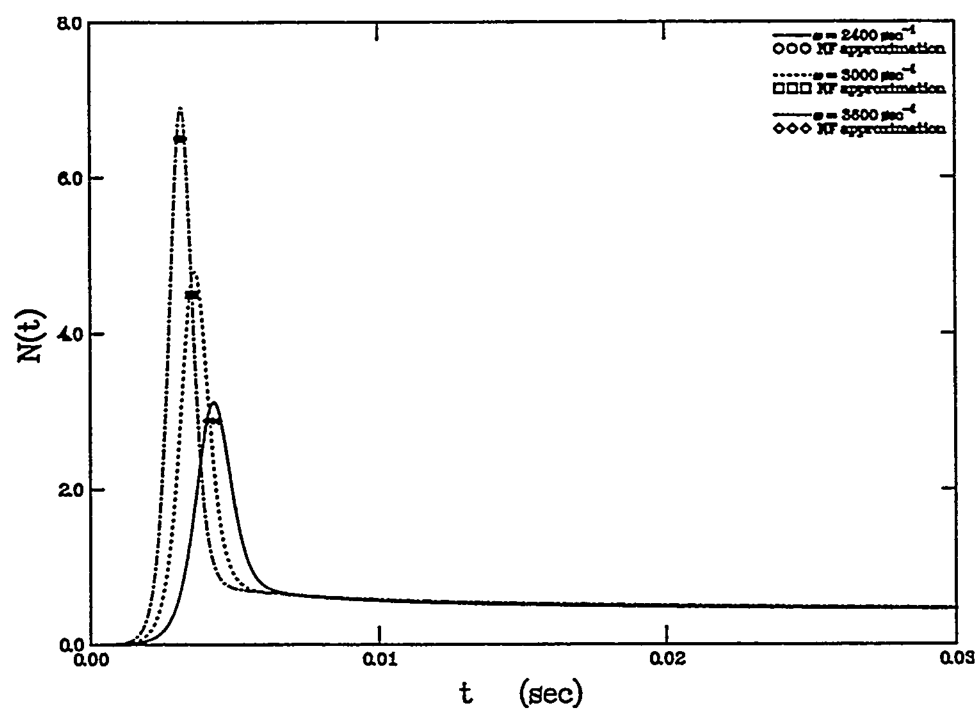


Figure F.1-2. Power histories (watts) of ^{235}U fast assembly and NF approximation.

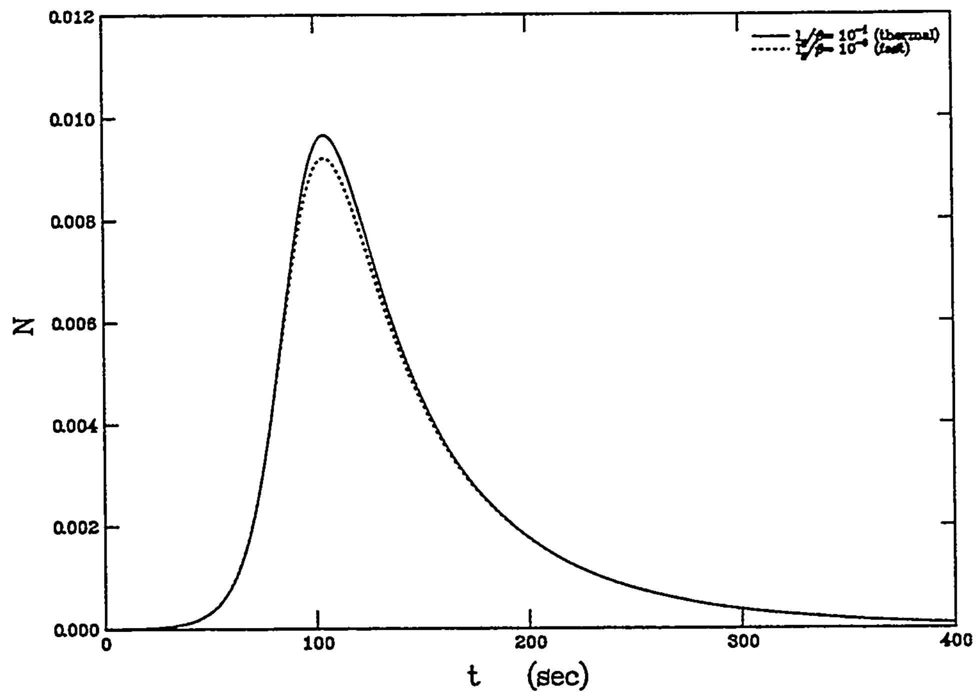


Figure F.2-1. Power histories (watts) of ^{235}U for small reactivity excursions.

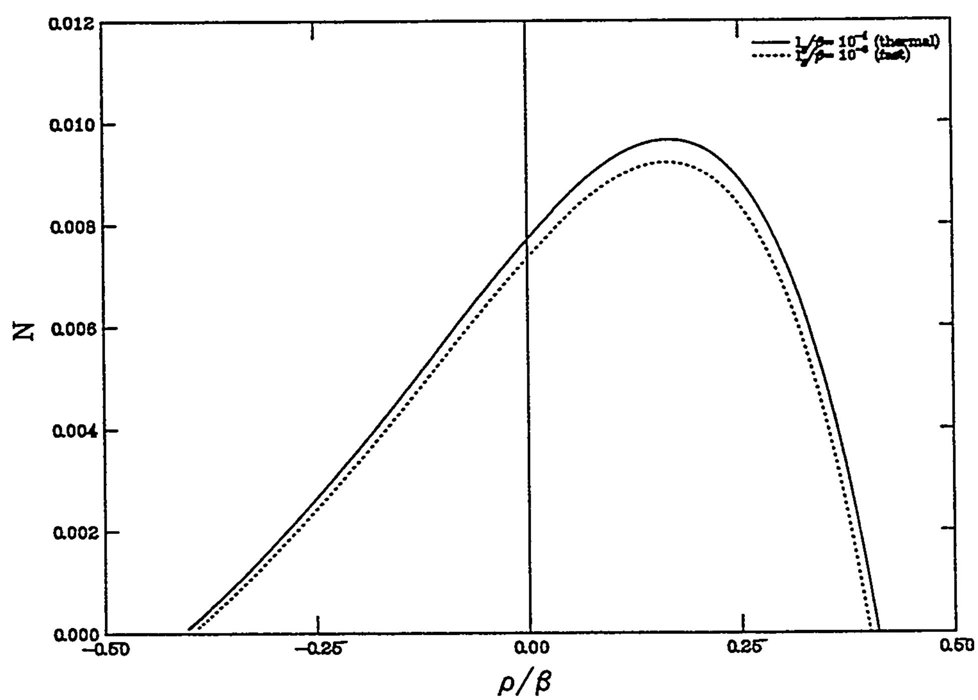


Figure F-4. Power (watts) versus reactivity of ^{235}U for small reactivity excursions.

Table F.2-1. NARK UDX Sensitivity Input Parameters

NARK Sensitivity Input Parameter	Units	Number of Sensitivity Input Parameters	Values of Sensitivity Input Parameters
TFM mass	Kg	3	25.0, 50.0, 100.0
TFM volume Fraction	Dimensionless	1	262.155×10^{-6}
Effective neutron lifetime, l	S	2	1.0×10^{-4}, 5.0×10^{-4}
Excess reactivity insertion, ρ_0(¢)	¢	3	0.1, 1.0, 10.0
Initial Power, N_0	W	3	0.001, 0.1, 1.0
Fuel-Doppler temperature coefficient, α_T^F	k_{eff}/k_{eff}/K	3	-7.5×10^{-5}, 10^{-4}, -2.5×10^{-4}

Table F.2-2. Key Input Parameters for NARK UDX Sensitivity Calculations

NARK RUN ID	Initial Power (w)	Initial Power (fissions/sec)	Initial Reactivity (cents)	TFM Mass (kg)	Doppler Coefficient (pcm)	Neutron life, (sec)
dxf0001	1.000E-03	3.286E+07	1.000E-01	2.500E+01	-5.000E+00	1.000E-04
dxf0002	1.000E-03	3.286E+07	1.000E-01	2.500E+01	-5.000E+00	5.000E-05
dxf0003	1.000E-03	3.286E+07	1.000E-01	2.500E+01	-1.000E+01	1.000E-04
dxf0004	1.000E-03	3.286E+07	1.000E-01	2.500E+01	-1.000E+01	5.000E-05
dxf0005	1.000E-03	3.286E+07	1.000E-01	2.500E+01	-5.000E+01	1.000E-04
dxf0006	1.000E-03	3.286E+07	1.000E-01	2.500E+01	-5.000E+01	5.000E-05
dxf0007	1.000E-03	3.286E+07	1.000E-01	5.000E+01	-5.000E+00	1.000E-04
dxf0008	1.000E-03	3.286E+07	1.000E-01	5.000E+01	-5.000E+00	5.000E-05
dxf0009	1.000E-03	3.286E+07	1.000E-01	5.000E+01	-1.000E+01	1.000E-04
dxf0010	1.000E-03	3.286E+07	1.000E-01	5.000E+01	-1.000E+01	5.000E-05
dxf0011	1.000E-03	3.286E+07	1.000E-01	5.000E+01	-5.000E+01	1.000E-04
dxf0012	1.000E-03	3.286E+07	1.000E-01	5.000E+01	-5.000E+01	5.000E-05
dxf0013	1.000E-03	3.286E+07	1.000E-01	1.000E+02	-5.000E+00	1.000E-04
dxf0014	1.000E-03	3.286E+07	1.000E-01	1.000E+02	-5.000E+00	5.000E-05
dxf0015	1.000E-03	3.286E+07	1.000E-01	1.000E+02	-1.000E+01	1.000E-04
dxf0016	1.000E-03	3.286E+07	1.000E-01	1.000E+02	-1.000E+01	5.000E-05
dxf0017	1.000E-03	3.286E+07	1.000E-01	1.000E+02	-5.000E+01	1.000E-04
dxf0018	1.000E-03	3.286E+07	1.000E-01	1.000E+02	-5.000E+01	5.000E-05
dxf0019	1.000E-03	3.286E+07	1.000E+00	2.500E+01	-5.000E+00	1.000E-04
dxf0020	1.000E-03	3.286E+07	1.000E+00	2.500E+01	-5.000E+00	5.000E-05
dxf0021	1.000E-03	3.286E+07	1.000E+00	2.500E+01	-1.000E+01	1.000E-04
dxf0022	1.000E-03	3.286E+07	1.000E+00	2.500E+01	-1.000E+01	5.000E-05
dxf0023	1.000E-03	3.286E+07	1.000E+00	2.500E+01	-5.000E+01	1.000E-04
dxf0024	1.000E-03	3.286E+07	1.000E+00	2.500E+01	-5.000E+01	5.000E-05
dxf0025	1.000E-03	3.286E+07	1.000E+00	5.000E+01	-5.000E+00	1.000E-04
dxf0026	1.000E-03	3.286E+07	1.000E+00	5.000E+01	-5.000E+00	5.000E-05
dxf0027	1.000E-03	3.286E+07	1.000E+00	5.000E+01	-1.000E+01	1.000E-04
dxf0028	1.000E-03	3.286E+07	1.000E+00	5.000E+01	-1.000E+01	5.000E-05
dxf0029	1.000E-03	3.286E+07	1.000E+00	5.000E+01	-5.000E+01	1.000E-04
dxf0030	1.000E-03	3.286E+07	1.000E+00	5.000E+01	-5.000E+01	5.000E-05
dxf0031	1.000E-03	3.286E+07	1.000E+00	1.000E+02	-5.000E+00	1.000E-04
dxf0032	1.000E-03	3.286E+07	1.000E+00	1.000E+02	-5.000E+00	5.000E-05
dxf0033	1.000E-03	3.286E+07	1.000E+00	1.000E+02	-1.000E+01	1.000E-04
dxf0034	1.000E-03	3.286E+07	1.000E+00	1.000E+02	-1.000E+01	5.000E-05
dxf0035	1.000E-03	3.286E+07	1.000E+00	1.000E+02	-5.000E+01	1.000E-04
dxf0036	1.000E-03	3.286E+07	1.000E+00	1.000E+02	-5.000E+01	5.000E-05
dxf0037	1.000E-03	3.286E+07	1.000E+01	2.500E+01	-5.000E+00	1.000E-04
dxf0038	1.000E-03	3.286E+07	1.000E+01	2.500E+01	-5.000E+00	5.000E-05
dxf0039	1.000E-03	3.286E+07	1.000E+01	2.500E+01	-1.000E+01	1.000E-04
dxf0040	1.000E-03	3.286E+07	1.000E+01	2.500E+01	-1.000E+01	5.000E-05
dxf0041	1.000E-03	3.286E+07	1.000E+01	2.500E+01	-5.000E+01	1.000E-04
dxf0042	1.000E-03	3.286E+07	1.000E+01	2.500E+01	-5.000E+01	5.000E-05
dxf0043	1.000E-03	3.286E+07	1.000E+01	5.000E+01	-5.000E+00	1.000E-04
dxf0044	1.000E-03	3.286E+07	1.000E+01	5.000E+01	-5.000E+00	5.000E-05
dxf0045	1.000E-03	3.286E+07	1.000E+01	5.000E+01	-1.000E+01	1.000E-04
dxf0046	1.000E-03	3.286E+07	1.000E+01	5.000E+01	-1.000E+01	5.000E-05
dxf0047	1.000E-03	3.286E+07	1.000E+01	5.000E+01	-5.000E+01	1.000E-04
dxf0048	1.000E-03	3.286E+07	1.000E+01	5.000E+01	-5.000E+01	5.000E-05
dxf0049	1.000E-03	3.286E+07	1.000E+01	1.000E+02	-5.000E+00	1.000E-04
dxf0050	1.000E-03	3.286E+07	1.000E+01	1.000E+02	-5.000E+00	5.000E-05
dxf0051	1.000E-03	3.286E+07	1.000E+01	1.000E+02	-1.000E+01	1.000E-04
dxf0052	1.000E-03	3.286E+07	1.000E+01	1.000E+02	-1.000E+01	5.000E-05
dxf0053	1.000E-03	3.286E+07	1.000E+01	1.000E+02	-5.000E+01	1.000E-04
dxf0054	1.000E-03	3.286E+07	1.000E+01	1.000E+02	-5.000E+01	5.000E-05
dxf0055	1.000E-01	3.286E+09	1.000E-01	2.500E+01	-5.000E+00	1.000E-04
dxf0056	1.000E-01	3.286E+09	1.000E-01	2.500E+01	-5.000E+00	5.000E-05

Table F.2-2. Key Input Parameters for NARK UDX Sensitivity Calculations (Continued)

dxf0057	1.000E-01	3.286E+09	1.000E-01	2.500E+01	-1.000E+01	1.000E-04
dxf0058	1.000E-01	3.286E+09	1.000E-01	2.500E+01	-1.000E+01	5.000E-05
dxf0059	1.000E-01	3.286E+09	1.000E-01	2.500E+01	-5.000E+01	1.000E-04
dxf0060	1.000E-01	3.286E+09	1.000E-01	2.500E+01	-5.000E+01	5.000E-05
dxf0061	1.000E-01	3.286E+09	1.000E-01	5.000E+01	-5.000E+00	1.000E-04
dxf0062	1.000E-01	3.286E+09	1.000E-01	5.000E+01	-5.000E+00	5.000E-05
dxf0063	1.000E-01	3.286E+09	1.000E-01	5.000E+01	-1.000E+01	1.000E-04
dxf0064	1.000E-01	3.286E+09	1.000E-01	5.000E+01	-1.000E+01	5.000E-05
dxf0065	1.000E-01	3.286E+09	1.000E-01	5.000E+01	-5.000E+01	1.000E-04
dxf0066	1.000E-01	3.286E+09	1.000E-01	5.000E+01	-5.000E+01	5.000E-05
dxf0067	1.000E-01	3.286E+09	1.000E-01	1.000E+02	-5.000E+00	1.000E-04
dxf0068	1.000E-01	3.286E+09	1.000E-01	1.000E+02	-5.000E+00	5.000E-05
dxf0069	1.000E-01	3.286E+09	1.000E-01	1.000E+02	-1.000E+01	1.000E-04
dxf0070	1.000E-01	3.286E+09	1.000E-01	1.000E+02	-1.000E+01	5.000E-05
dxf0071	1.000E-01	3.286E+09	1.000E-01	1.000E+02	-5.000E+01	1.000E-04
dxf0072	1.000E-01	3.286E+09	1.000E-01	1.000E+02	-5.000E+01	5.000E-05
dxf0073	1.000E-01	3.286E+09	1.000E+00	2.500E+01	-5.000E+00	1.000E-04
dxf0074	1.000E-01	3.286E+09	1.000E+00	2.500E+01	-5.000E+00	5.000E-05
dxf0075	1.000E-01	3.286E+09	1.000E+00	2.500E+01	-1.000E+01	1.000E-04
dxf0076	1.000E-01	3.286E+09	1.000E+00	2.500E+01	-1.000E+01	5.000E-05
dxf0077	1.000E-01	3.286E+09	1.000E+00	2.500E+01	-5.000E+01	1.000E-04
dxf0078	1.000E-01	3.286E+09	1.000E+00	2.500E+01	-5.000E+01	5.000E-05
dxf0079	1.000E-01	3.286E+09	1.000E+00	5.000E+01	-5.000E+00	1.000E-04
dxf0080	1.000E-01	3.286E+09	1.000E+00	5.000E+01	-5.000E+00	5.000E-05
dxf0081	1.000E-01	3.286E+09	1.000E+00	5.000E+01	-1.000E+01	1.000E-04
dxf0082	1.000E-01	3.286E+09	1.000E+00	5.000E+01	-1.000E+01	5.000E-05
dxf0083	1.000E-01	3.286E+09	1.000E+00	5.000E+01	-5.000E+01	1.000E-04
dxf0084	1.000E-01	3.286E+09	1.000E+00	5.000E+01	-5.000E+01	5.000E-05
dxf0085	1.000E-01	3.286E+09	1.000E+00	1.000E+02	-5.000E+00	1.000E-04
dxf0086	1.000E-01	3.286E+09	1.000E+00	1.000E+02	-5.000E+00	5.000E-05
dxf0087	1.000E-01	3.286E+09	1.000E+00	1.000E+02	-1.000E+01	1.000E-04
dxf0088	1.000E-01	3.286E+09	1.000E+00	1.000E+02	-1.000E+01	5.000E-05
dxf0089	1.000E-01	3.286E+09	1.000E+00	1.000E+02	-5.000E+01	1.000E-04
dxf0090	1.000E-01	3.286E+09	1.000E+00	1.000E+02	-5.000E+01	5.000E-05
dxf0091	1.000E-01	3.286E+09	1.000E+01	2.500E+01	-5.000E+00	1.000E-04
dxf0092	1.000E-01	3.286E+09	1.000E+01	2.500E+01	-5.000E+00	5.000E-05
dxf0093	1.000E-01	3.286E+09	1.000E+01	2.500E+01	-1.000E+01	1.000E-04
dxf0094	1.000E-01	3.286E+09	1.000E+01	2.500E+01	-1.000E+01	5.000E-05
dxf0095	1.000E-01	3.286E+09	1.000E+01	2.500E+01	-5.000E+01	1.000E-04
dxf0096	1.000E-01	3.286E+09	1.000E+01	2.500E+01	-5.000E+01	5.000E-05
dxf0097	1.000E-01	3.286E+09	1.000E+01	5.000E+01	-5.000E+00	1.000E-04
dxf0098	1.000E-01	3.286E+09	1.000E+01	5.000E+01	-5.000E+00	5.000E-05
dxf0099	1.000E-01	3.286E+09	1.000E+01	5.000E+01	-1.000E+01	1.000E-04
dxf0100	1.000E-01	3.286E+09	1.000E+01	5.000E+01	-1.000E+01	5.000E-05
dxf0101	1.000E-01	3.286E+09	1.000E+01	5.000E+01	-5.000E+01	1.000E-04
dxf0102	1.000E-01	3.286E+09	1.000E+01	5.000E+01	-5.000E+01	5.000E-05
dxf0103	1.000E-01	3.286E+09	1.000E+01	1.000E+02	-5.000E+00	1.000E-04
dxf0104	1.000E-01	3.286E+09	1.000E+01	1.000E+02	-5.000E+00	5.000E-05
dxf0105	1.000E-01	3.286E+09	1.000E+01	1.000E+02	-1.000E+01	1.000E-04
dxf0106	1.000E-01	3.286E+09	1.000E+01	1.000E+02	-1.000E+01	5.000E-05
dxf0107	1.000E-01	3.286E+09	1.000E+01	1.000E+02	-5.000E+01	1.000E-04
dxf0108	1.000E-01	3.286E+09	1.000E+01	1.000E+02	-5.000E+01	5.000E-05
dxf0109	1.000E+00	3.286E+10	1.000E-01	2.500E+01	-5.000E+00	1.000E-04
dxf0110	1.000E+00	3.286E+10	1.000E-01	2.500E+01	-5.000E+00	5.000E-05
dxf0111	1.000E+00	3.286E+10	1.000E-01	2.500E+01	-1.000E+01	1.000E-04
dxf0112	1.000E+00	3.286E+10	1.000E-01	2.500E+01	-1.000E+01	5.000E-05
dxf0113	1.000E+00	3.286E+10	1.000E-01	2.500E+01	-5.000E+01	1.000E-04
dxf0114	1.000E+00	3.286E+10	1.000E-01	2.500E+01	-5.000E+01	5.000E-05

Table F.2-2. Key Input Parameters for NARK UDX Sensitivity Calculations (Continued)

dxf0115	1.000E+00	3.286E+10	1.000E-01	5.000E+01	-5.000E+00	1.000E-04
dxf0116	1.000E+00	3.286E+10	1.000E-01	5.000E+01	-5.000E+00	5.000E-05
dxf0117	1.000E+00	3.286E+10	1.000E-01	5.000E+01	-1.000E+01	1.000E-04
dxf0118	1.000E+00	3.286E+10	1.000E-01	5.000E+01	-1.000E+01	5.000E-05
dxf0119	1.000E+00	3.286E+10	1.000E-01	5.000E+01	-5.000E+01	1.000E-04
dxf0120	1.000E+00	3.286E+10	1.000E-01	5.000E+01	-5.000E+01	5.000E-05
dxf0121	1.000E+00	3.286E+10	1.000E-01	1.000E+02	-5.000E+00	1.000E-04
dxf0122	1.000E+00	3.286E+10	1.000E-01	1.000E+02	-5.000E+00	5.000E-05
dxf0123	1.000E+00	3.286E+10	1.000E-01	1.000E+02	-1.000E+01	1.000E-04
dxf0124	1.000E+00	3.286E+10	1.000E-01	1.000E+02	-1.000E+01	5.000E-05
dxf0125	1.000E+00	3.286E+10	1.000E-01	1.000E+02	-5.000E+01	1.000E-04
dxf0126	1.000E+00	3.286E+10	1.000E-01	1.000E+02	-5.000E+01	5.000E-05
dxf0127	1.000E+00	3.286E+10	1.000E+00	2.500E+01	-5.000E+00	1.000E-04
dxf0128	1.000E+00	3.286E+10	1.000E+00	2.500E+01	-5.000E+00	5.000E-05
dxf0129	1.000E+00	3.286E+10	1.000E+00	2.500E+01	-1.000E+01	1.000E-04
dxf0130	1.000E+00	3.286E+10	1.000E+00	2.500E+01	-1.000E+01	5.000E-05
dxf0131	1.000E+00	3.286E+10	1.000E+00	2.500E+01	-5.000E+01	1.000E-04
dxf0132	1.000E+00	3.286E+10	1.000E+00	2.500E+01	-5.000E+01	5.000E-05
dxf0133	1.000E+00	3.286E+10	1.000E+00	5.000E+01	-5.000E+00	1.000E-04
dxf0134	1.000E+00	3.286E+10	1.000E+00	5.000E+01	-5.000E+00	5.000E-05
dxf0135	1.000E+00	3.286E+10	1.000E+00	5.000E+01	-1.000E+01	1.000E-04
dxf0136	1.000E+00	3.286E+10	1.000E+00	5.000E+01	-1.000E+01	5.000E-05
dxf0137	1.000E+00	3.286E+10	1.000E+00	5.000E+01	-5.000E+01	1.000E-04
dxf0138	1.000E+00	3.286E+10	1.000E+00	5.000E+01	-5.000E+01	5.000E-05
dxf0139	1.000E+00	3.286E+10	1.000E+00	1.000E+02	-5.000E+00	1.000E-04
dxf0140	1.000E+00	3.286E+10	1.000E+00	1.000E+02	-5.000E+00	5.000E-05
dxf0141	1.000E+00	3.286E+10	1.000E+00	1.000E+02	-1.000E+01	1.000E-04
dxf0142	1.000E+00	3.286E+10	1.000E+00	1.000E+02	-1.000E+01	5.000E-05
dxf0143	1.000E+00	3.286E+10	1.000E+00	1.000E+02	-5.000E+01	1.000E-04
dxf0144	1.000E+00	3.286E+10	1.000E+00	1.000E+02	-5.000E+01	5.000E-05
dxf0145	1.000E+00	3.286E+10	1.000E+01	2.500E+01	-5.000E+00	1.000E-04
dxf0146	1.000E+00	3.286E+10	1.000E+01	2.500E+01	-5.000E+00	5.000E-05
dxf0147	1.000E+00	3.286E+10	1.000E+01	2.500E+01	-1.000E+01	1.000E-04
dxf0148	1.000E+00	3.286E+10	1.000E+01	2.500E+01	-1.000E+01	5.000E-05
dxf0149	1.000E+00	3.286E+10	1.000E+01	2.500E+01	-5.000E+01	1.000E-04
dxf0150	1.000E+00	3.286E+10	1.000E+01	2.500E+01	-5.000E+01	5.000E-05
dxf0151	1.000E+00	3.286E+10	1.000E+01	5.000E+01	-5.000E+00	1.000E-04
dxf0152	1.000E+00	3.286E+10	1.000E+01	5.000E+01	-5.000E+00	5.000E-05
dxf0153	1.000E+00	3.286E+10	1.000E+01	5.000E+01	-1.000E+01	1.000E-04
dxf0154	1.000E+00	3.286E+10	1.000E+01	5.000E+01	-1.000E+01	5.000E-05
dxf0155	1.000E+00	3.286E+10	1.000E+01	5.000E+01	-5.000E+01	1.000E-04
dxf0156	1.000E+00	3.286E+10	1.000E+01	5.000E+01	-5.000E+01	5.000E-05
dxf0157	1.000E+00	3.286E+10	1.000E+01	1.000E+02	-5.000E+00	1.000E-04
dxf0158	1.000E+00	3.286E+10	1.000E+01	1.000E+02	-5.000E+00	5.000E-05
dxf0159	1.000E+00	3.286E+10	1.000E+01	1.000E+02	-1.000E+01	1.000E-04
dxf0160	1.000E+00	3.286E+10	1.000E+01	1.000E+02	-1.000E+01	5.000E-05
dxf0161	1.000E+00	3.286E+10	1.000E+01	1.000E+02	-5.000E+01	1.000E-04
dxf0162	1.000E+00	3.286E+10	1.000E+01	1.000E+02	-5.000E+01	5.000E-05

Table F.2-3. Uncoupled Nuclear Dynamics Fixed Input Parameters

```
!1
!PRENARK 1.01 (08/20/97)                                26-AUG-1997 16:07:13
!======================================================================
!PRENARK  "FIXED INPUT PARAMETERS" TRANSFER FILE:
!POSTNARK_UDXFUE.FIP
!CODE VERSION: 1.01
!CODE REVISION DATE: 08/20/97
!EXECUTION MACHINE: BEATLE-ALPHA AXP
!OPERATING SYSTEM:  OpenVMS V6.1
!CODE AUTHORS:
!JONATHAN SCOTT RATH, ESQ. [UNM/NMERI]
!CODE SPONSORS:
!JONATHAN SCOTT RATH, ESQ. [UNM/NMERI]
!RUN DATE: 26-AUG-1997
!RUN TIME: 16:07:13
!======================================================================
!*VARIABLE....(VARIABLE_DESCRIPTION_              )......[UNITS]
! NARK_VALUE(S)
!======================================================================
*FUEL.........(THERMAL_FISSILE_MATERIAL_SPECIES  )......[NOT/APPLICABLE]
U235
*QN...........(NEUTRON_SOURCE_RATE               )......[W/SEC]
 0.0000000D+00
*FISDNS.......(THERMAL_FISSILE_MATERIAL_DENSITY  )......[KG/M^3]
 1.9070000D+04
*FISSPH.......(TFM_SPECIFIC_HEAT_CAPACITY        )......[J/KG/K]
 1.1600000D+02
*XDENSI.......(COMPOSITE_TFM+HOST_ROCK_DENSITY   )......[KG/M^3]
 2.2774040D+03
*XHECAP.......(COMPOSITE_TFM+HOST_ROCK_SPCHEATCAP)......[J/KG/K]
 1.3223457D+03
*FIVOFR.......(TFM_VOLUME/HOST_ROCK_VOLUME_RATIO )......[DIMENSIONLESS]
 2.6219190D-04
*FIMAFR.......(TFM_MASS/HOST_ROCK_MASS_RATIO     )......[DIMENSIONLESS]
 2.1954820D-03
*SSNF.........(NEUTRONS_PER_ENERGY_CONVERSION    )......[NEUTRONS/JOULE]
 7.9900000D+10
*FISPEN.......(ENERGY_PER_FISSION_CONVERSION     )......[FISSIONS/JOULE]
 3.2860000D+10
*BETATO.......(TOTAL_FRACTION_OF_DELAYED_NEUTRONS)......[DIMENSIONLESS]
 6.5000000D-03
*GAMBET.......(DELAYED_NEUTRON_FRAC+ENERGY_CORREC)......[DIMENSIONLESS]
 0.0000000D+00
*NG...........(NUMBER_OF_DELAYED_NEUTRON_GROUPS  )......[DIMENSIONLESS]
  7
*BETA.........(DELAYED_NEUTRON_FRACTIONS         )......[DIMENSIONLESS]
 0.0000000D+00
 2.1450000D-04
 1.4235000D-03
 1.2740000D-03
 2.5675000D-03
 7.4750000D-04
 2.7300000D-04
*LN(2)/LAMBDA.(DELAYED_NEUTRON_HALF_LIVES        )......[SEC]
 6.9314718D-04
 5.5898966D+01
 2.2726137D+01
 6.2445692D+00
 2.3028146D+00
 6.0802384D-01
 2.3028146D-01
*AVGDKY.......(AVERAGE_NEUTRON_DECAY_CONSTANT    )......[SEC^-1]
```

Table F.2-3. Uncoupled Nuclear Dynamics Fixed Input Parameters (Continued.)

```
 7.6719434D-02
*AVGNLF.......(AVERAGE_NEUTRON_LIFETIME          )......[SEC]
 8.4934295D-02
*NH...........(NO._OF_NEUTRON_DECAY_HEAT_GROUPS  )......[DIMENSIONLESS]
 11
*BETAH........(NEUTRON_DECAY_HEAT_FRACTIONS      )......[DIMENSIONLESS]
 2.9900000D-03
 8.2500000D-03
 1.5500000D-02
 1.9350000D-02
 1.1650000D-02
 6.4500000D-03
 2.3100000D-03
 1.6400000D-03
 8.5000000D-04
 4.3000000D-04
 5.7000000D-04
*LAMBDH.......(DECAY_HEAT_PRECURSOR_CONSTANTS    )......[SEC^-1]
 1.7720000D+00
 5.7740000D-01
 6.7430000D-02
 6.2140000D-03
 4.7390000D-04
 4.8100000D-05
 5.3440000D-06
 5.7260000D-07
 1.0360000D-07
 2.9590000D-08
 7.5850000D-10
*BETAHT......(TOTAL_NEUTRON_DECAY_HEAT_FRACTION  )......[DIMENSIONLESS]
 6.9990000D-02
*END_FIXED_INPUT_PARAMETERS_FILE
!==================================================================
```

Table F.2-4 Key Results from NARK UDX Sensitivity Calculations

NARK Run ID	Excursion Time [sec]	Max Avg Flux [n/cm2/s]	Max Temp Rise [K]	Peak Power [w]	Peak Power [fis/sec]	Total Energy [fission]
dxf0001	3.118E+05	1.905E+08	2.792E-01	8.555E+01	2.811E+12	1.381E+17
dxf0002	3.116E+05	1.907E+08	2.793E-01	8.567E+01	2.815E+12	1.382E+17
dxf0003	2.948E+05	9.507E+07	1.395E-01	4.269E+01	1.403E+12	6.904E+16
dxf0004	2.946E+05	9.518E+07	1.396E-01	4.274E+01	1.404E+12	6.905E+16
dxf0005	2.559E+05	1.900E+07	2.790E-02	8.533E+00	2.804E+11	1.380E+16
dxf0006	2.559E+05	1.900E+07	2.788E-02	8.533E+00	2.804E+11	1.380E+16
dxf0007	3.288E+05	1.901E+08	2.791E-01	1.708E+02	5.612E+12	2.762E+17
dxf0008	3.286E+05	1.902E+08	2.792E-01	1.709E+02	5.616E+12	2.762E+17
dxf0009	3.119E+05	9.530E+07	1.396E-01	8.560E+01	2.813E+12	1.381E+17
dxf0010	3.116E+05	9.534E+07	1.396E-01	8.564E+01	2.814E+12	1.381E+17
dxf0011	2.727E+05	1.901E+07	2.791E-02	1.708E+01	5.612E+11	2.762E+16
dxf0012	2.726E+05	1.903E+07	2.791E-02	1.709E+01	5.617E+11	2.761E+16
dxf0013	3.455E+05	1.891E+08	2.791E-01	3.399E+02	1.117E+13	5.525E+17
dxf0014	3.453E+05	1.891E+08	2.791E-01	3.399E+02	1.117E+13	5.524E+17
dxf0015	3.288E+05	9.508E+07	1.395E-01	1.708E+02	5.614E+12	2.762E+17
dxf0016	3.286E+05	9.510E+07	1.396E-01	1.709E+02	5.615E+12	2.762E+17
dxf0017	2.895E+05	1.892E+07	2.791E-02	3.400E+01	1.117E+12	5.523E+16
dxf0018	2.893E+05	1.892E+07	2.791E-02	3.400E+01	1.117E+12	5.524E+16
dxf0019	4.239E+04	1.912E+10	2.792E+00	8.518E+03	2.799E+14	1.381E+18
dxf0020	4.238E+04	1.913E+10	2.791E+00	8.521E+03	2.800E+14	1.381E+18
dxf0021	4.071E+04	9.557E+09	1.396E+00	4.256E+03	1.399E+14	6.906E+17
dxf0022	4.069E+04	9.564E+09	1.396E+00	4.259E+03	1.400E+14	6.905E+17
dxf0023	3.679E+04	1.913E+09	2.791E-01	8.521E+02	2.800E+13	1.381E+17
dxf0024	3.676E+04	1.914E+09	2.791E-01	8.525E+02	2.801E+13	1.381E+17
dxf0025	4.409E+04	1.902E+10	2.791E+00	1.695E+04	5.571E+14	2.762E+18
dxf0026	4.406E+04	1.902E+10	2.791E+00	1.695E+04	5.571E+14	2.762E+18
dxf0027	4.240E+04	9.561E+09	1.396E+00	8.519E+03	2.799E+14	1.381E+18
dxf0028	4.237E+04	9.568E+09	1.396E+00	8.526E+03	2.802E+14	1.381E+18
dxf0029	3.846E+04	1.909E+09	2.791E-01	1.701E+03	5.590E+13	2.762E+17
dxf0030	3.844E+04	1.909E+09	2.791E-01	1.701E+03	5.591E+13	2.762E+17
dxf0031	4.579E+04	1.902E+10	2.791E+00	3.386E+04	1.113E+15	5.525E+18
dxf0032	4.577E+04	1.904E+10	2.791E+00	3.390E+04	1.114E+15	5.524E+18
dxf0033	4.409E+04	9.505E+09	1.396E+00	1.695E+04	5.569E+14	2.762E+18
dxf0034	4.405E+04	9.514E+09	1.396E+00	1.696E+04	5.574E+14	2.763E+18
dxf0035	4.016E+04	1.897E+09	2.792E-01	3.383E+03	1.112E+14	5.526E+17
dxf0036	4.014E+04	1.896E+09	2.791E-01	3.382E+03	1.111E+14	5.525E+17
dxf0037	5.502E+03	2.054E+12	2.800E+01	9.012E+05	2.961E+16	1.385E+19
dxf0038	5.499E+03	2.057E+12	2.799E+01	9.026E+05	2.966E+16	1.385E+19
dxf0039	5.329E+03	1.030E+12	1.399E+01	4.524E+05	1.487E+16	6.925E+18
dxf0040	5.325E+03	1.031E+12	1.400E+01	4.528E+05	1.488E+16	6.926E+18
dxf0041	4.918E+03	2.055E+11	2.800E+00	9.029E+04	2.967E+15	1.385E+18
dxf0042	4.916E+03	2.053E+11	2.799E+00	9.023E+04	2.965E+15	1.385E+18
dxf0043	5.681E+03	2.049E+12	2.799E+01	1.802E+06	5.920E+16	2.770E+19
dxf0044	5.677E+03	2.047E+12	2.799E+01	1.800E+06	5.914E+16	2.770E+19
dxf0045	5.503E+03	1.026E+12	1.400E+01	9.007E+05	2.960E+16	1.385E+19
dxf0046	5.499E+03	1.028E+12	1.400E+01	9.026E+05	2.966E+16	1.385E+19
dxf0047	5.094E+03	2.060E+11	2.799E+00	1.809E+05	5.944E+15	2.770E+18
dxf0048	5.091E+03	2.062E+11	2.800E+00	1.811E+05	5.950E+15	2.770E+18
dxf0049	5.855E+03	2.062E+12	2.800E+01	3.622E+06	1.190E+17	5.541E+19
dxf0050	5.853E+03	2.062E+12	2.799E+01	3.622E+06	1.190E+17	5.540E+19
dxf0051	5.680E+03	1.025E+12	1.400E+01	1.802E+06	5.922E+16	2.770E+19
dxf0052	5.678E+03	1.023E+12	1.400E+01	1.799E+06	5.913E+16	2.770E+19
dxf0053	5.271E+03	2.038E+11	2.799E+00	3.576E+05	1.175E+16	5.541E+18
dxf0054	5.268E+03	2.043E+11	2.799E+00	3.584E+05	1.178E+16	5.540E+18
dxf0055	2.042E+05	1.736E+08	2.721E-01	7.797E+01	2.562E+12	1.346E+17
dxf0056	2.041E+05	1.737E+08	2.722E-01	7.801E+01	2.564E+12	1.346E+17

Table F.2-4 Key Results from NARK UDX Sensitivity Calculations (Continued)

dxf0057	1.888E+05	8.414E+07	1.339E-01	3.779E+01	1.242E+12	6.621E+16
dxf0058	1.888E+05	8.419E+07	1.339E-01	3.782E+01	1.243E+12	6.621E+16
dxf0059	1.481E+05	1.701E+07	2.632E-02	7.641E+00	2.511E+11	1.298E+16
dxf0060	1.480E+05	1.702E+07	2.632E-02	7.645E+00	2.512E+11	1.298E+16
dxf0061	2.191E+05	1.828E+08	2.759E-01	1.642E+02	5.396E+12	2.730E+17
dxf0062	2.190E+05	1.830E+08	2.759E-01	1.643E+02	5.400E+12	2.730E+17
dxf0063	2.042E+05	8.681E+07	1.361E-01	7.797E+01	2.562E+12	1.347E+17
dxf0064	2.041E+05	8.685E+07	1.361E-01	7.801E+01	2.564E+12	1.346E+17
dxf0065	1.663E+05	1.689E+07	2.640E-02	1.518E+01	4.988E+11	2.608E+16
dxf0066	1.662E+05	1.690E+07	2.640E-02	1.519E+01	4.991E+11	2.608E+16
dxf0067	2.345E+05	1.872E+08	2.779E-01	3.364E+02	1.105E+13	5.499E+17
dxf0068	2.344E+05	1.872E+08	2.779E-01	3.365E+02	1.106E+13	5.500E+17
dxf0069	2.191E+05	9.142E+07	1.380E-01	1.643E+02	5.397E+12	2.730E+17
dxf0070	2.190E+05	9.140E+07	1.379E-01	1.642E+02	5.396E+12	2.730E+17
dxf0071	1.836E+05	1.684E+07	2.666E-02	3.025E+01	9.942E+11	5.271E+16
dxf0072	1.835E+05	1.685E+07	2.666E-02	3.027E+01	9.947E+11	5.272E+16
dxf0073	3.115E+04	1.910E+10	2.791E+00	8.506E+03	2.795E+14	1.381E+18
dxf0074	3.113E+04	1.912E+10	2.792E+00	8.514E+03	2.798E+14	1.381E+18
dxf0075	2.945E+04	9.491E+09	1.396E+00	4.225E+03	1.388E+14	6.906E+17
dxf0076	2.944E+04	9.497E+09	1.396E+00	4.227E+03	1.389E+14	6.905E+17
dxf0077	2.557E+04	1.892E+09	2.788E-01	8.422E+02	2.767E+13	1.380E+17
dxf0078	2.556E+04	1.893E+09	2.788E-01	8.427E+02	2.769E+13	1.379E+17
dxf0079	3.285E+04	1.913E+10	2.792E+00	1.704E+04	5.601E+14	2.763E+18
dxf0080	3.283E+04	1.913E+10	2.791E+00	1.704E+04	5.601E+14	2.762E+18
dxf0081	3.115E+04	9.554E+09	1.396E+00	8.508E+03	2.796E+14	1.381E+18
dxf0082	3.113E+04	9.558E+09	1.396E+00	8.513E+03	2.797E+14	1.381E+18
dxf0083	2.724E+04	1.910E+09	2.791E-01	1.701E+03	5.590E+13	2.762E+17
dxf0084	2.724E+04	1.910E+09	2.790E-01	1.701E+03	5.589E+13	2.761E+17
dxf0085	3.452E+04	1.906E+10	2.791E+00	3.398E+04	1.116E+15	5.524E+18
dxf0086	3.450E+04	1.905E+10	2.791E+00	3.396E+04	1.116E+15	5.524E+18
dxf0087	3.286E+04	9.560E+09	1.395E+00	1.703E+04	5.598E+14	2.761E+18
dxf0088	3.283E+04	9.568E+09	1.396E+00	1.705E+04	5.603E+14	2.762E+18
dxf0089	2.892E+04	1.911E+09	2.791E-01	3.406E+03	1.119E+14	5.523E+17
dxf0090	2.890E+04	1.912E+09	2.792E-01	3.408E+03	1.120E+14	5.525E+17
dxf0091	4.333E+03	2.057E+12	2.799E+01	9.027E+05	2.966E+16	1.385E+19
dxf0092	4.330E+03	2.061E+12	2.800E+01	9.045E+05	2.972E+16	1.385E+19
dxf0093	4.155E+03	1.029E+12	1.400E+01	4.521E+05	1.486E+16	6.925E+18
dxf0094	4.153E+03	1.029E+12	1.400E+01	4.521E+05	1.486E+16	6.925E+18
dxf0095	3.746E+03	2.049E+11	2.799E+00	9.006E+04	2.959E+15	1.385E+18
dxf0096	3.745E+03	2.047E+11	2.799E+00	8.999E+04	2.957E+15	1.385E+18
dxf0097	4.508E+03	2.042E+12	2.800E+01	1.796E+06	5.902E+16	2.770E+19
dxf0098	4.505E+03	2.039E+12	2.799E+01	1.794E+06	5.894E+16	2.770E+19
dxf0099	4.332E+03	1.029E+12	1.400E+01	9.031E+05	2.968E+16	1.385E+19
dxf0100	4.330E+03	1.030E+12	1.400E+01	9.044E+05	2.972E+16	1.385E+19
dxf0101	3.923E+03	2.061E+11	2.800E+00	1.810E+05	5.947E+15	2.771E+18
dxf0102	3.921E+03	2.063E+11	2.800E+00	1.812E+05	5.954E+15	2.771E+18
dxf0103	4.684E+03	2.062E+12	2.800E+01	3.622E+06	1.190E+17	5.541E+19
dxf0104	4.682E+03	2.062E+12	2.800E+01	3.623E+06	1.190E+17	5.541E+19
dxf0105	4.507E+03	1.021E+12	1.400E+01	1.795E+06	5.900E+16	2.771E+19
dxf0106	4.505E+03	1.020E+12	1.400E+01	1.794E+06	5.896E+16	2.770E+19
dxf0107	4.098E+03	2.046E+11	2.799E+00	3.590E+05	1.180E+16	5.540E+18
dxf0108	4.096E+03	2.049E+11	2.799E+00	3.596E+05	1.182E+16	5.540E+18
dxf0109	1.481E+05	1.701E+08	2.632E-01	7.641E+01	2.511E+12	1.298E+17
dxf0110	1.480E+05	1.702E+08	2.632E-01	7.645E+01	2.512E+12	1.298E+17
dxf0111	1.292E+05	8.616E+07	1.318E-01	3.871E+01	1.272E+12	6.480E+16
dxf0112	1.292E+05	8.621E+07	1.318E-01	3.873E+01	1.273E+12	6.480E+16
dxf0113	8.471E+04	1.903E+07	2.703E-02	8.553E+00	2.811E+11	1.296E+16
dxf0114	8.467E+04	1.904E+07	2.703E-02	8.557E+00	2.812E+11	1.296E+16
dxf0115	1.663E+05	1.689E+08	2.640E-01	1.518E+02	4.988E+12	2.608E+17
dxf0116	1.662E+05	1.690E+08	2.640E-01	1.519E+02	4.991E+12	2.608E+17
dxf0117	1.481E+05	8.503E+07	1.316E-01	7.640E+01	2.511E+12	1.298E+17
dxf0118	1.480E+05	8.508E+07	1.316E-01	7.645E+01	2.512E+12	1.298E+17

Table F.2-4 Key Results from NARK UDX Sensitivity Calculations (Continued)

dxf0119	1.041E+05	1.791E+07	2.661E-02	1.609E+01	5.288E+11	2.592E+16
dxf0120	1.041E+05	1.792E+07	2.661E-02	1.610E+01	5.291E+11	2.591E+16
dxf0121	1.836E+05	1.684E+08	2.666E-01	3.025E+02	9.942E+12	5.271E+17
dxf0122	1.835E+05	1.685E+08	2.666E-01	3.027E+02	9.947E+12	5.272E+17
dxf0123	1.663E+05	8.447E+07	1.320E-01	1.518E+02	4.988E+12	2.608E+17
dxf0124	1.662E+05	8.453E+07	1.320E-01	1.519E+02	4.991E+12	2.608E+17
dxf0125	1.232E+05	1.734E+07	2.640E-02	3.117E+01	1.024E+12	5.183E+16
dxf0126	1.232E+05	1.735E+07	2.640E-02	3.119E+01	1.025E+12	5.183E+16
dxf0127	2.557E+04	1.892E+10	2.788E+00	8.422E+03	2.767E+14	1.380E+18
dxf0128	2.556E+04	1.893E+10	2.788E+00	8.427E+03	2.769E+14	1.379E+18
dxf0129	2.396E+04	9.400E+09	1.390E+00	4.190E+03	1.377E+14	6.877E+17
dxf0130	2.394E+04	9.405E+09	1.390E+00	4.192E+03	1.377E+14	6.879E+17
dxf0131	2.044E+04	1.720E+09	2.711E-01	7.667E+02	2.520E+13	1.341E+17
dxf0132	2.044E+04	1.720E+09	2.710E-01	7.668E+02	2.520E+13	1.341E+17
dxf0133	2.724E+04	1.910E+10	2.791E+00	1.701E+04	5.590E+14	2.762E+18
dxf0134	2.724E+04	1.910E+10	2.790E+00	1.701E+04	5.589E+14	2.761E+18
dxf0135	2.557E+04	9.460E+09	1.394E+00	8.422E+03	2.768E+14	1.379E+18
dxf0136	2.556E+04	9.463E+09	1.394E+00	8.425E+03	2.768E+14	1.379E+18
dxf0137	2.193E+04	1.798E+09	2.751E-01	1.604E+03	5.270E+13	2.722E+17
dxf0138	2.193E+04	1.798E+09	2.750E-01	1.603E+03	5.269E+13	2.721E+17
dxf0139	2.892E+04	1.911E+10	2.791E+00	3.406E+04	1.119E+15	5.523E+18
dxf0140	2.890E+04	1.912E+10	2.792E+00	3.408E+04	1.120E+15	5.525E+18
dxf0141	2.725E+04	9.545E+09	1.395E+00	1.700E+04	5.587E+14	2.761E+18
dxf0142	2.723E+04	9.560E+09	1.396E+00	1.703E+04	5.596E+14	2.762E+18
dxf0143	2.345E+04	1.873E+09	2.776E-01	3.336E+03	1.096E+14	5.494E+17
dxf0144	2.344E+04	1.874E+09	2.776E-01	3.338E+03	1.097E+14	5.494E+17
dxf0145	3.746E+03	2.049E+12	2.799E+01	9.006E+05	2.959E+16	1.385E+19
dxf0146	3.745E+03	2.047E+12	2.799E+01	8.999E+05	2.957E+16	1.385E+19
dxf0147	3.571E+03	1.026E+12	1.400E+01	4.501E+05	1.479E+16	6.926E+18
dxf0148	3.569E+03	1.027E+12	1.400E+01	4.509E+05	1.482E+16	6.925E+18
dxf0149	3.168E+03	2.048E+11	2.795E+00	8.987E+04	2.953E+15	1.383E+18
dxf0150	3.166E+03	2.048E+11	2.795E+00	8.990E+04	2.954E+15	1.383E+18
dxf0151	3.923E+03	2.061E+12	2.800E+01	1.810E+06	5.947E+16	2.771E+19
dxf0152	3.921E+03	2.063E+12	2.800E+01	1.812E+06	5.954E+16	2.771E+19
dxf0153	3.746E+03	1.024E+12	1.400E+01	9.004E+05	2.959E+16	1.385E+19
dxf0154	3.744E+03	1.024E+12	1.400E+01	9.003E+05	2.958E+16	1.385E+19
dxf0155	3.340E+03	2.033E+11	2.798E+00	1.788E+05	5.877E+15	2.769E+18
dxf0156	3.338E+03	2.035E+11	2.799E+00	1.788E+05	5.875E+15	2.770E+18
dxf0157	4.098E+03	2.046E+12	2.799E+01	3.590E+06	1.180E+17	5.540E+19
dxf0158	4.096E+03	2.049E+12	2.799E+01	3.596E+06	1.182E+17	5.540E+19
dxf0159	3.924E+03	1.030E+12	1.400E+01	1.810E+06	5.946E+16	2.770E+19
dxf0160	3.921E+03	1.031E+12	1.400E+01	1.811E+06	5.952E+16	2.770E+19
dxf0161	3.515E+03	2.058E+11	2.799E+00	3.617E+05	1.189E+16	5.539E+18
dxf0162	3.512E+03	2.058E+11	2.799E+00	3.617E+05	1.189E+16	5.540E+18

APPENDIX G
Thermal-Hydrology (THX) Model Features

The BRAGFLO_T code, used in the THX model, is a transient multi-phase (water and vapor) fluid and energy simulator. It has been used extensively in repository performance assessments in its current form (Rechard, 1995b) and its isothermal form (WIPP 1992)a. The enhancements include the addition of the energy balance equation and the incorporation of thermal effects on both fluid and rock properties. The code also contains submodels that predict gas and water consumption/production as a result of waste package corrosion, and a submodel that predicts the energy released as the result of radioactive decay of the waste. BRAGFLO_T also includes many features and solution techniques used in TOUGH2 (Pruess, 1991), such as effective continuum approximation for modeling fractured porous media, vapor pressure lowering due to capillary pressure and diffusive mass flux in the gas phase.

The BRAGFLO_T code uses a finite difference formulation to solve the coupled partial differential equations (PDEs) that describe the mass and energy balance of a two-component, two-phase system. Fick's law and a multi-phase extension of Darcy's Law are used to describe the fluid flow. Heat is transported by conduction and convection, the latter including both sensible and latent heat. . The following governing partial differential equations solved by BRAGFLO_T:

Component mass balance equations (i=1,2):

$$
\begin{aligned}
&\nabla\cdot\left[\frac{\alpha y_i \rho_n k_{rn} k}{\mu_n}\left(\nabla P_n + \rho_n g \nabla Z\right)\right] + \\
&\nabla\cdot\left[\frac{\alpha x_i \rho_w k_{rw} k}{\mu_w}\left(\nabla P_w + \rho_w g \nabla Z\right)\right] + \\
&\nabla\cdot\left[\frac{\alpha \rho_n \phi S_n D_{ij}}{\tau}\nabla y_i\right] + \\
&\alpha\left[y_i^s q_n^s + x_i^s q_w^s + y_i^c q_n^c + x_i^c q_w^c\right] \\
&= \alpha\frac{\partial}{\partial t}\left[y_i \phi \rho_n S_n + x_i \phi \rho_w S_w\right]
\end{aligned}
\qquad \text{(G-1)}
$$

Energy balance equation:

$$
\begin{aligned}
&\nabla\cdot(\alpha\lambda\nabla T)+\\
&\nabla\cdot\left[\frac{\alpha h_n\rho_n k_{rn}k}{\mu_n}(\nabla P_n+\rho_n g\nabla Z)\right]+\\
&\nabla\cdot\left[\frac{\alpha h_w\rho_w k_{rw}k}{\mu_w}(\nabla P_w+\rho_w g\nabla Z)\right]+\\
&\alpha\left[h_n^s q_n^s+h_w^s q_w^s+h_n^c q_n^c+h_w^c q_w^c\right]\\
&=\alpha\frac{\partial}{\partial t}\left[\phi(U_n\rho_n S_n+U_w\rho_w S_w)\right]+\alpha\frac{\partial}{\partial t}\left[(1-\phi)\rho_r U_r\right]
\end{aligned}
\qquad \text{(G-2)}
$$

In the component mass balance equation, the mass must be compositional with respect to i, where i is equal to 1 for the noncondensable component (air), and 2 for the condensable component (water). Subscripts w and n refer to the wetting and nonwetting phases, respectively, while subscript r refers to the rock. Superscripts are used to distinguish the origin of a source/sink term; s refers to wells, and c to chemical (and nuclear) reactions.

The power output of the event(s) was assumed to be a function of both space and time, which then can be defined as a power density relation. For the THX calculations, two different space power density functions (SPDF's) were used:

$$
N_H(r,t)=N_{H,0}H\left[1-\frac{t}{\tau}\right]\cos\left(\frac{\pi}{2r_0}r\right) \qquad \text{(G-3)}
$$

$$
N_S(r,t)=N_{S,0}\sin\left(\frac{\pi}{\tau}t\right)\cos\left(\frac{\pi}{2r_0}r\right) \qquad \text{(G-4)}
$$

where N_H and N_S represent the Heaviside and sinusoidal power density functions, $N_{H,0}$ and $N_{S,0}$ are the power density constants [J/(s•m^3)], τ is the power generation period [s], and r_0 is the radius of the heat generation zone (HGZ) [m].

The size of the critical events described here ranged from 2 x 10^{18} to 1.5 x 10^{20} fissions per event. (Trivial thermal pulses were obtained at values less than the minimum.) The total energy released, E_0, for a given event is related to the number of fissions by the conversion factor, 1.0 fission $\cong$ 30.4 x 10^{-12} J. The power density constants (normalization parameters), $N_{H,0}$ and $N_{S,0}$ were found by integrating the above space power density functions over the spherical space domain (r=0 to r=r_0) and in time (t=0 to t=τ), yielding,

$$
N_{H,0}=\frac{E}{4\ \pi\ \tau\ r_0^3\left[\left(\frac{2}{\pi}\right)-\left(\frac{16}{\pi^3}\right)\right]} \qquad \text{(G-5)}
$$

$$N_{s,o} = \frac{E}{8\ \tau\ r_0^3 \left[\left(\frac{2}{\pi} \right) - \left(\frac{16}{\pi^3} \right) \right]} \quad \text{(G-6)}$$

The THX model consisted of a 1-dimensional (1-D) sphere containing a mixture of host rock (volcanic tuff), water, and fissile material. These calculations involved two different types (or profiles) of excursion pulses, three different heat generation zone radii, and several different total input integrated fissions (energy). The computational results yielded the time behavior for power and geological repository responses (*i.e.*, saturation, thermal, flow histories). A total of 45 THX calculations were completed using the experimental version of BRAGFLO_T in the NDCA study. Knowledge of the saturation and temperature histories allowed the corresponding "recycle" times to be computed. The "recycle" time is simply the time period for some dependent variable (*i.e.*, saturation or temperature) to return to the initial condition. These saturation and temperature "recycle" times were then used to identify integrated fissions resulting from geologic repository criticalities.

Symbols:

D_{ij}	gas diffusion coefficient of component *i* in a mixture of *i* and *j* [m^2/s]
E	energy [J]
N	power density function [J/(m^3•s)
P	pressure [Pa]
S	saturation [dimensionless]
T	temperature [K]
U	internal energy [J]
Z	elevation [m]
g	gravitational acceleration [m/s^2]
h	enthalpy [J/kg]
k	permeability [m^2]
k_{eff}	neutron multiplication factor [dimensionless]
k_r	relative permeability [dimensionless]
q	source/sink term [kg/(m^3•s)]
r	radius [m]
t	time [s]
x_i	mass fraction of component *i* in the wetting phase [dimensionless]
y_I	mass fraction of component *i* in the nonwetting phase [dimensionless
α	dimensionless-dependent geometry term
α	[(1D=$\Delta y \Delta z$, m^2),(2D=Δz, m),(3D=1, dimensionless)]
∇	del (gradient) operator [dimensionless]
τ	power generation period [s]
ρ	density [kg/m^3]
ϕ	porosity [dimensionless]

G.1 Discretization

A sphere of welded tuff, 500 m in diameter (see diagram in Figure G.1-1), containing a spherical heat-generating zone (due to the critical assembly) that is either $r_0 = 0.5$, 1.0, or 1.5 m in radius, was simulated using 200 one-dimensional (radial) finite difference grid blocks. The geometry was modeled as a hemisphere (by symmetry) and was discretized in the radial direction such that the each grid block contained a geometrical volume that is at most 1.5 times larger than the next grid block volume to the interior. The spherical discretization was fine enough that each grid block volume was at most 0.75 times the sum of all interior grid block volumes. Starting with an interior radius of 0.05 m, 200 grid blocks were used to model the 500 meter radius sphere, of which the first 30 grid blocks were used to contain the heat generation zone. The spherical geometry is approximated using Cartesian coordinates; hence, element thickness (ΔZ) values were computed as a function of the internal and external radius of the grid block to conserve volume. Incorporating symmetry into the spherical geometry model, only 2π steradians of the sphere were necessary, yielding the relationship

$$\Delta Z = \frac{\frac{2}{3}\pi\left(r_0^3 - r_i^3\right)}{\Delta X} \qquad \text{(G.1-1)}$$

where r_0 is the external radius of an individual grid block, r_i is the internal radius of an individual grid block, and ΔX is equal to r_0 - r_i. Thus, the volume of the grid block is equal to $\Delta X \bullet \Delta Y \bullet \Delta Z$, where $\Delta Y = 1.0$. The detailed grid block dimensions for each of the unique r_0 (radius of the spherical heat generation zone) finite difference grid (or mesh) are displayed in Figure G.1-1.

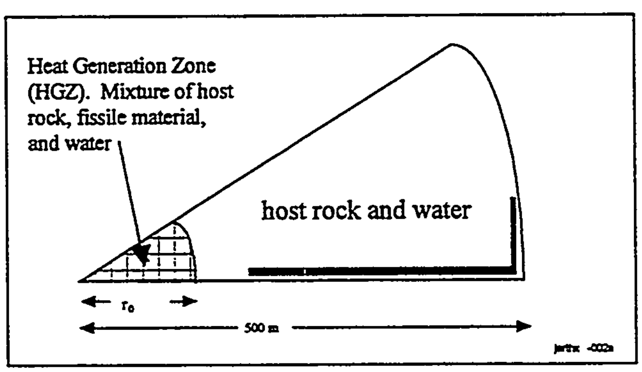


Figure G.1-1. HGZ assembly zone used in the THX calculations.

G.2 Initial and Boundary Conditions

The welded tuff was assumed to have several different initial saturation values of 65%, 75%, and 85%. The remaining void space in the pores was assumed to be filled with gas (air). To impose these conditions the BRAGFLO_T calculations were begun with: an initial water pressure of -1.20477×10^6 Pa, -7.57500×10^5 Pa, and -3.98000×10^5 Pa (corresponding to the 65%, 75% and 85% saturation values, respectively), an initial gas (air) pressure of 1.1×10^5 Pa, and an initial temperature of 299.15 K. (The negative initial water pressures are a result of the capillary pressure function, $P_c = P_w - P_n$. Using the capillary pressure function vs. saturation, the initial saturation can then be determined (Rechard, 1995b).)

No flow (*i.e.*, Neumann type) boundary conditions were imposed on the mass flux at both ends of the problem domain. Since heat was assumed to be transported only from the surface of the sphere, the boundary heat loss option of BRAGFLO_T was used for the extreme boundary condition ($r = r_0$). At the center of the sphere, $r = 0$, a Neumann boundary condition was imposed on the energy flux. Results from the THX model can be found in Section 5.3.

APPENDIX H
PRA Model (SLAM)

(Rough Order of Magnitude Analysis of the Probabilities for FEPs Resulting in a Critical Event)

H.1 Overview

This appendix presents net findings from a preliminary probabilistic risk assessment (PRA) used to estimate rough order of magnitude (ROM) values for the probabilities of nuclear criticality for spent nuclear fuel disposed in the Yucca Mountain repository. This PRA approach was used at the present time because the two alternative methods for obtaining criticality probabilities, "Hybrid Abstraction" and a Particle-In-a-Cell (PIC, a code that includes all applicable physics necessary to model criticality initators) code were not available. At the present time, the modeling of criticality scenarios within the Yucca Mountain repository have not reached a high level of maturity. The set of system submodels important to nuclear criticality (corrosion allowance material – outer canister material, corrosion resistant material – inner low corrosion rate canister material, fuel cladding, etc., have not all been appropriately modeled to the extent that probabilities for criticality can be determined accurately. There is a good confidence in the existing PRA methodology, however, there isn't enough resolution in current PA models, especially those corrosion, to adequately determine waste package product probabilities that could lead to a criticality. In order to get a fundamental basis for the probability of critical events, a rough order of magnitude (ROM) analysis was performed using a PRA approach using the simulation code "SLAM". Detail discussion of the PRA/SLAM model is not provided here since the PRA model is currently being upgraded and the SLAM code will be replaced with the SAPHIRE code. Updated results will be published in the near future by the INEEL participants in the NDCA project.

The PRA/SLAM results, using assumed upper limit probabilities for modeling parameters, identified that criticality may occur in the in situ or the near-field over a 100,000-year period (only under high water infiltration rates) and is very unlikely in the far-field. Far-field criticalities are very difficult to generate since reconcentrations of transported fissile material would require very exceptional conditions to occur. The preliminary PRA presented a ROM estimate of approximately two in situ or near-field criticalities occurring per 100,000-year period when experiencing high water infiltration periods (i.e. glacial conditions). These criticalities are not of major concern because they would result in only a very small increase in the fission yield products. The quantity (magnitude) of these additional fission yield products is so small that when these values are integrated into the source term radionuclide inventory, their contribution would be orders of magnitude less than source term inventory values at time of repository closure (see Sections 6 and 7).

It is important for the reader to understand that even though preliminary criticality analysis results (probabilities) are presented in this appendix in event tree/fault tree format, the results were obtained by using a simulation code. In general, event tree/fault tree PRA methodology are not capable of modeling the detailed performance of geologic repository and predict probabilities for conditional scenarios leading to criticality. The reason is that many of the event probability distributions are a function of time and have conditionally dependent event-timing. However time-dependent conditional scenarios can be modeled in Monte-Carlo simulation codes (in the time domain). The calculation of the distribution for the time-to-failure values of the waste packages were done with the simulation code "SLAM", with a total of 313,100 trials (100 per DOE package) run in the Monte-Carlo simulations. Post-processing of these simulations was then conducted to estimate generalized probabilities for key events. Key DOE-owned fuel features are identified in Table H.1-1, H.1-2 and H1-3. The results are present in appendix sections H.2 and for H.3 for short- and long-term infiltration rates.

Table H.1-1. Fuel Type and Degraded Configuration Subcritical Mass (kg) Limits (a)

Fuel Type (b)	Unpoisoned (c) Structure	Collapsed (d) Fuel Rods	Slurry (e)
ATR - Aluminum	Safe	NA	Safe (>3% poison) 14.4 kg (1% poison) 10 kg (no poison)
N-Reactor	Safe	Safe	Safe (<200 kg nonstratified) 150 kg (stratified)
Shippingport	Safe	Safe	NA
Ft. St. Vrain	Safe	Safe	NA
Triga Flip	Safe	NA	Safe (>3% poison) 14.4 kg (1% poison) 10 kg (no poison)
Triga Standard	Safe	NA	Safe
Westinghouse PWR	See RW (f)		

(a) For this analysis, $k_{eff} < .95$ was assumed to be subcritical because this is a probabilistic analysis, not a worst-case deterministic one.
(b) See Tables C.1-1 and C.1-2 and Rechard 1997 for more details of fuel types.
(c) Unpoisoned structure corresponds to degraded *in situ* geometry with loss of boron but iron oxide supports the intact fuel elements.
(d) Collapsed fuel rod configuration corresponds to corroded and collapsed borated stainless steel supports and collapsed, but intact, fuel elements.
(e) Slurry configuration corresponds to a geometry where all components have lost integrity and structure. Some fuel types are categorized by the remaining percentage of initial package neutron poison content.
(f) See Ref. RW 1997.

Table H.1-2. Number of Potentially Critical Packages per Fuel/Waste Type (a)

Fuel Category	Fuel Type	No. of Packages	No. of Pkgs Containing ^{235}U (b) with a Mass Greater Than Credible Critical Mass	Fraction of DOE (c) Packages at Critical Mass (3,131 Pkgs)	No. of Pkgs Containing (d) ^{235}U With a Mass Greater Than Criticality Cutoff
1	N-Reactor	115	100 (>216 kg)	0.03	100
2	HWCTR	7	NA	-	NA
3	Fermi	70	NA	-	NA
4	PBF	417	NA	-	NA
5	TMI	361	NA	-	NA
6	ATR, MTR	504	380 (>10 kg)	0.12	17
7	FRR	26	NA	-	NA
8	Ft. St. Vrain	139	0	0	0
9	Peach Bottom	26	NA	-	NA
10	FFTF	5	NA	-	NA
11	EBRII	123	NA	-	NA
12	Shippingport	65	0	0	0
13	Triga Standard Triga FLIPP	76 26	7 (>10 kg)	0.0036	2
14	Commercial (PWR)	4,820	See RW (e)	-	See RW (e)
15	Commercial (BWR)	2,859	See RW (e)	-	See RW (e)
16	HLW glass	11,353	0	0	0
TOTAL	DOE SNF	1,960			
	All SNF	9,639			
	All Packages	20,992			

(a) See Tables C.1-1 and C.1-2 for more details of fuel type.
(b) Number of packages with mass greater than potential (only for slurry degraded configuration with major loss of initial poison material, see Table H.1-1) credible critical mass.
(c) Fractional representation of column 4 values (based on total number of DOE SNF and HLW packages = 3,131.)
(d) Based on criticality mass cutoff.
(e) See Ref. RW 1997.

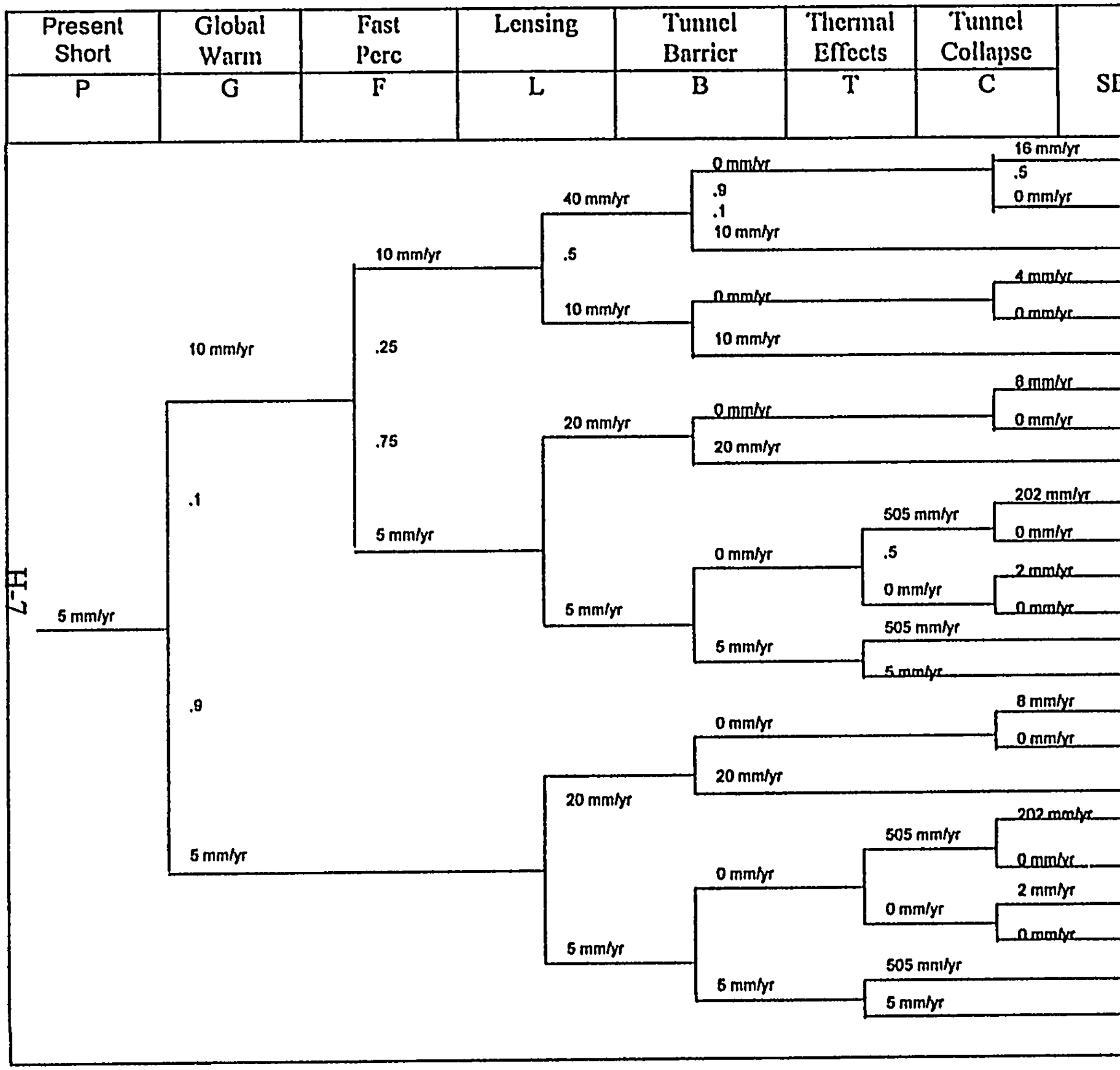
Table H.1-3. Major Models Containing Component Events Leading to Possible Criticality Configurations.

Model and Component Description	Scenario Short Term (S) Long Term (L)	Attributes Probability (P) Rate (R) Duration (D)
Water Infiltration Model		
Present infiltration mode	S/L	PRD
Global warming effects	S	PRD
Fast percolation effect	S/L	PRD
Lensing	S/L	PR
Tunnel Barrier effects	S/L	PRD
Geothermal effects	S	PRD
Glacial weather effects	L	PRD
Glacial local max	L	PR
Worst case glacial onset	L	PRD
Package/Fuel Degradation Model		
Cask corrodes	S/L	PD
Support structure fails	S/L	P
Fuel structure fails	S/L	PD
Boron removal/cask flooding	S	PD
Criticality Configuration Model		
In-situ-slurry at base	S/L	PD
In-situ matrix intact	S/L	PD
Near-Field Fuel dissolution	S/L	PD
Far-Field -Fuel migration	S/L	PD

H.2 Short-Term Infiltration Events Leading to Criticality

The event tree in Figure H.2-1 corresponds to groundwater infiltration due to short-term events which are expected to occur over the first 300 to 500 years. The dominate phenomena during this time may be expected to include: (1) short-term infiltration, (2) global warming, (3) reflux (flow back of thermally transported pore water), (4) thermal halo (the thermally induced drying of host rock adjacent to waste package drifts), and (5) tunnel barrier (also known as the French Drain Scenario where thermally induced stress fractures around the drifts lead to a capillary barrier that diverts water around the tunnel, further discussion can be found in Appendix K). The key feature of the short term event tree analysis is the infiltration rate for which distribution values are tentative and are expected to change as new information becomes available. The distribution values for the short-term infiltration are computed in Figure H.2-1. From this figure it can be seen that the output infiltration values are listed on the tree for tracking purposes, they do not tie directly into probability. Probabilities at each level of infiltration are presented in Figure H.2-2.

Figure H.2-1 Short-Term Infiltration Model

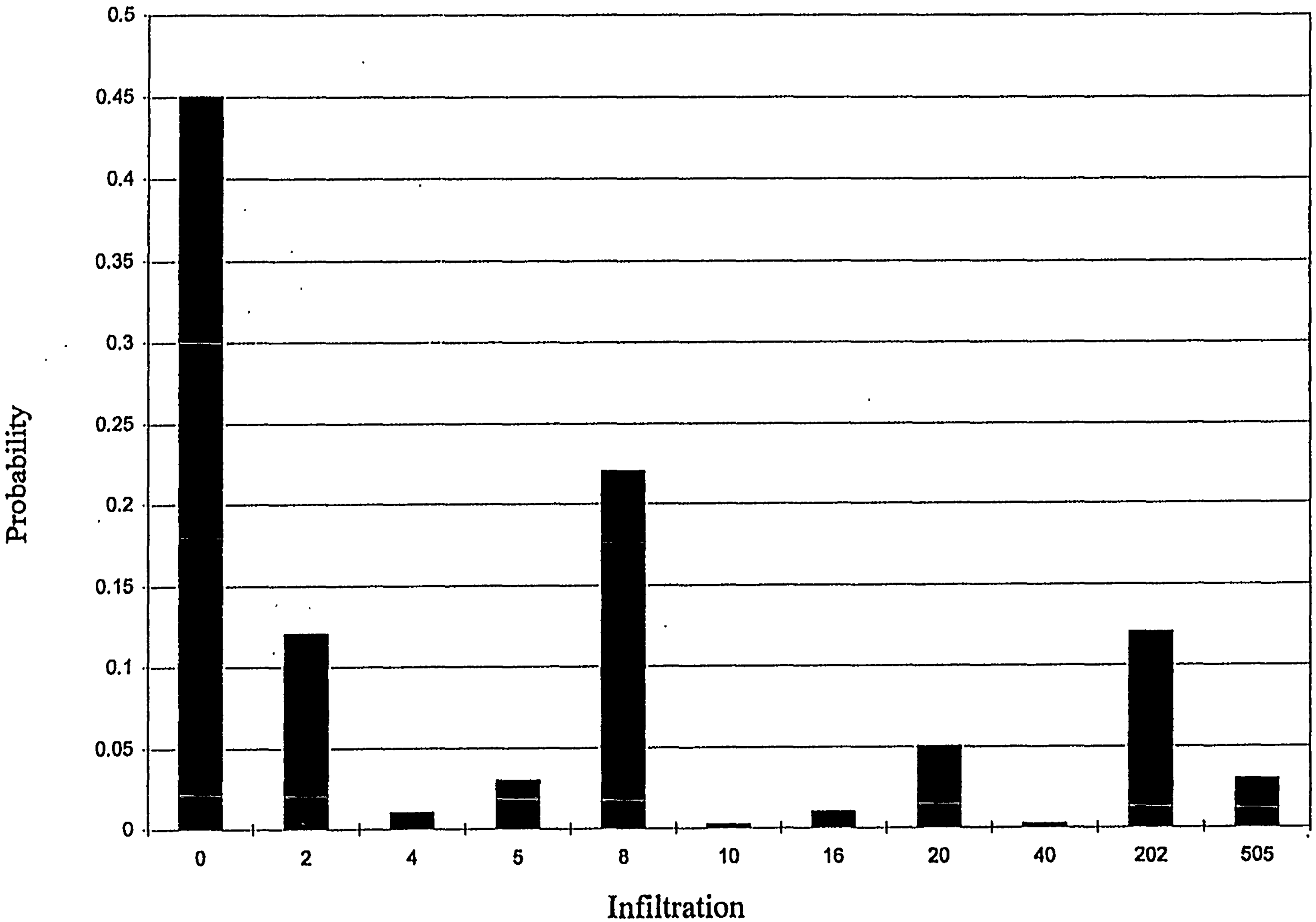


SEQ #	END-STATE-NAMES	FREQUENCY
1	HALO-Q/BAR-COLL	0.0056
2	HALO-Q/BARRIER	0.0056
3	HALO-QUENCHED	0.0013
4	HALO-Q/BAR-COLL	0.0056
5	HALO-Q/BARRIER	0.0056
6	HALO-Q/NO-BARRIER	0.0013
7	HALO-Q/BAR-COLL	0.0169
8	HALO-Q/BARRIER	0.0169
9	HALO-QUENCHED	0.0038
10	HALO/BAR-COLL	0.0084
11	BARRIER	0.0084
12	NO-THERMAL/BAR-COLL	0.0084
13	NO-THERMAL/BARRIER	0.0084
14	NO-BARRIER	0.0019
15	NO-THERMAL-FX	0.0019
16	HALO-Q/BAR-COLL	0.2025
17	HALO-Q/BARRIER	0.2025
18	HALO-QUENCHED	0.0450
19	THERMAL-FX/BARRIER	0.1013
20	THERMAL-FX/BARRIER	0.1013
21	NO-THERMAL/BAR-COLL	0.1013
22	NO-THERMAL/BARRIER	0.1013
23	THERMAL-FX/NO-BARRIER	0.0225
24	NO-THERMAL/NO-BARRIER	0.0225

HALO - thermal effects are possible at infiltration rate < 7 mm/yr.
HALO-Q - thermal effects quenched by infiltration > 7 mm/yr.
BARRIER - tunnel can act as a capillary barrier.
BAR-COLL - tunnel acts as a barrier, but has collapsed.
NO-BARRIER - tunnel does not act as a capillary barrier.

THERMAL-FX - reflux of in-pore water can occur
NO-THERMAL - thermal effects are not possible even at infiltration rate < 7 mm/yr.

Figure H.2-2. Computed Distribution for Short-Term Infiltration Events (a)

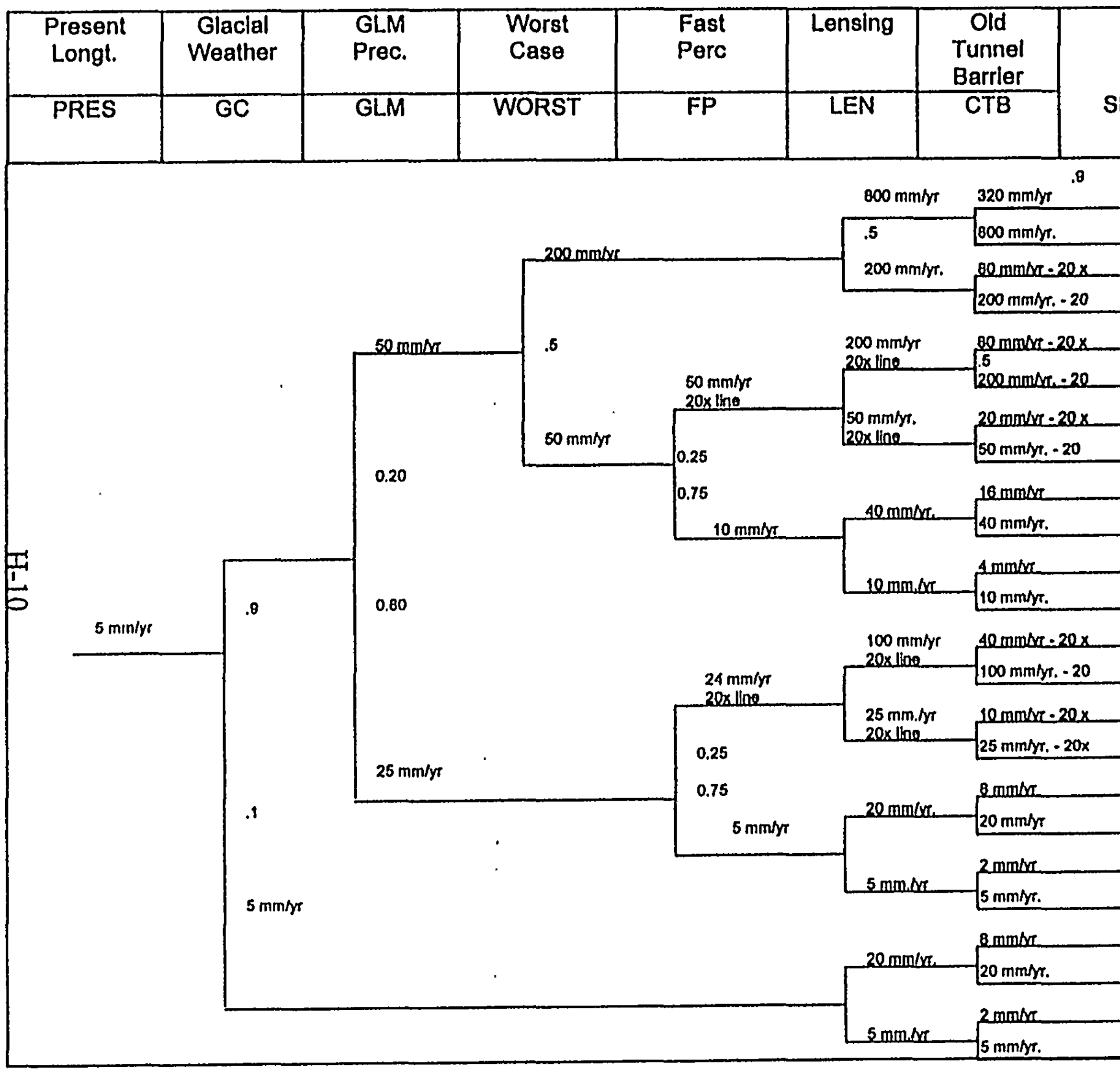


(a) see Figure H.2-1 for event tree which is used to calculate the distribution for short-term infiltration values.

H.3 Long-Term Infiltration Events Leading to Criticality

The event tree in Figure H.3-1 corresponds to groundwater infiltration due to long-term climatic events which are expected to occur during the next 25,000 years if glacial conditions exist. The dominate phenomena during the time includes: (1) present long-term infiltration, (2) glacial weather, (3) glacial local maximum precipitation, (4) "worst case" precipitation (essentially assuming that the next set of ice ages starts within a few hundred years and results in an increase in precipitation for 80% in the next 100,000 years, (5) fast percolation, (6) lensing, and (7) old tunnel barrier (capillary barrier). The key feature of the long-term event tree analysis is the large infiltration rate distribution. The distribution values for the long-term infiltration are computed in Figure H.3-1. Probabilities at each level of infiltration are presented in Figure H.3-2.

Figure H.3-1 Long-Term Infiltration Model

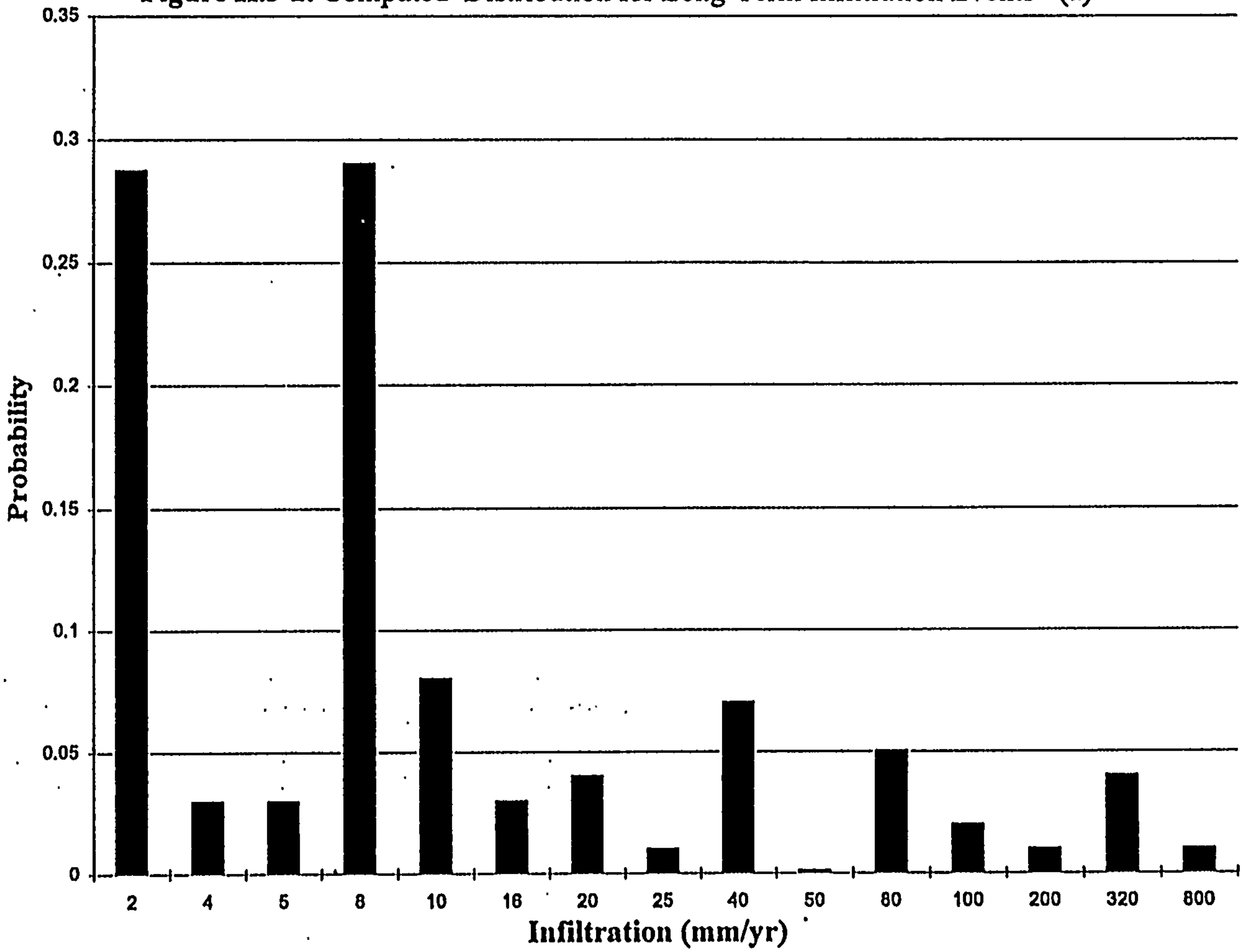


SEQ #	END-STATE-NAMES	FREQUENCY
1	GWORSTLB%P	0.0405
2	GWORSTL%P	0.0045
3	GWORSTB	0.0405
4	GWORST	0.0045
5	GGFLB%T%P	0.0101
6	GGFL%T%P	0.0011
7	CGFB%T	0.0101
8	GGF%T	0.0011
9	GGLB%P	0.0304
10	GGL%P	0.0034
11	GGB	0.0304
12	GGLATIAL	0.0034
13	GFLB%T%P	0.0810
14	GFL%T%P	0.0090
15	GFB%T	0.0810
16	GF%T	0.0090
17	GLB%P	0.2430
18	GL%P	0.0243
19	GB	0.2430
20	GLATIAL	0.0243
21	PLB%P	0.0450
22	PL%P	0.0050
23	PB	0.0450
24	PRESENT	0.0050

1st G - glacial conditions occur.
2nd G - GLM level of precipitation occurs.
WORST - worst-case level of precipitation occurs.
L - Lensing (absence of any letter - - event did not occur).
B - old tunnel acts as a limited capillary barrier.

%T - infiltration experienced only 20% of time.
F - "FAST PERC" conditions occur.
%P - small % of 20,900 packages experience in-pore moisture recycled by thermal halo.
P or PRESENT - "Present" levels of precipitation (5mm/yr) for entire time period

Figure H.3-2. Computed Distribution for Long-Term Infiltration Events (a)



(a) see Figure H.3-1 for event tree which is used to calculate the distribution for long-term infiltration values.