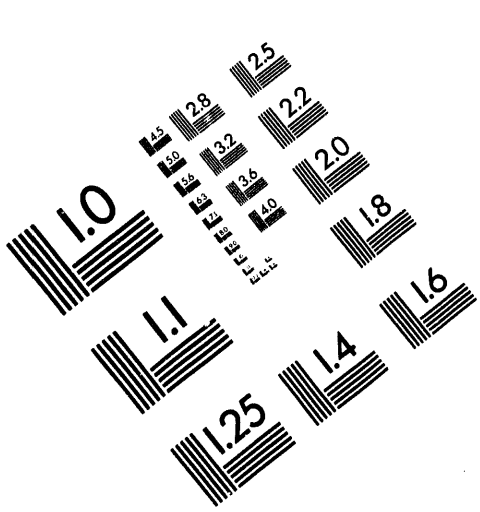
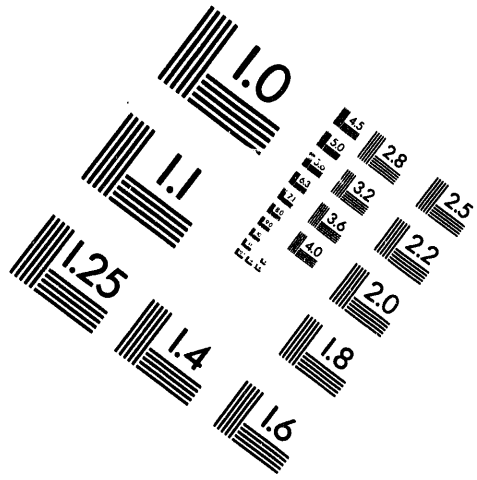
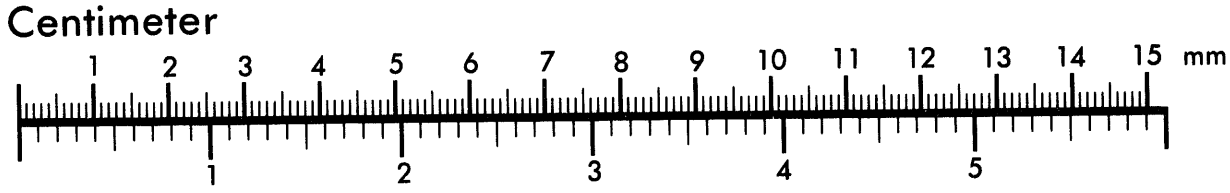


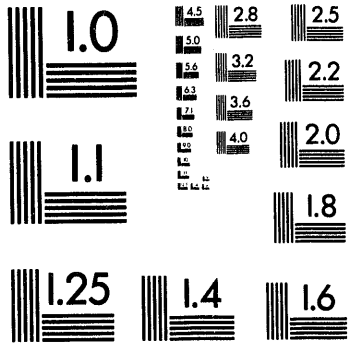
**Association for Information and Image Management**

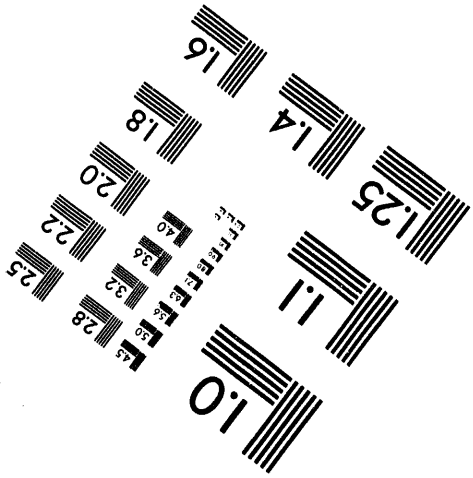
1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910

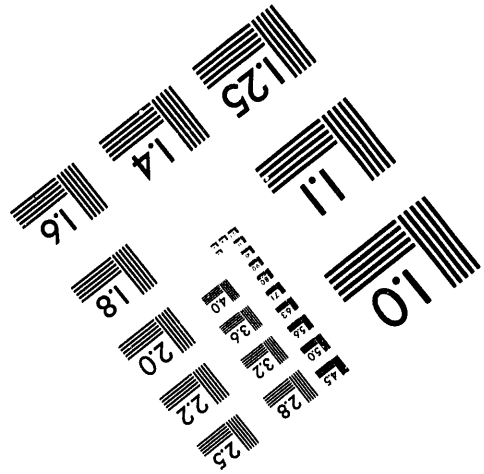
301/587-8202









1 of 1

# SUPPORTING DOCUMENT

1. Total Pages 72 75

| 2. Title | 3. Number | 4. Rev No. |
|---|---|---|
| Accident Safety Analysis for 300 Area N Reactor Fuel Fabrication and Storage Facility | WHC-SD-NR-RA-003 | 0 |

5. Key Words

Hazards identification, Determination of Hazard and Safety Classes, Abnormal Operations/Events and Accident Analysis

APPROVED FOR PUBLIC RELEASE
D. Birkland 1/24/94

6. Author

Name: D. J. Johnson, Restoration and Remediation Safety Analysis

D. J. Johnson
Signature

Organization/Charge Code 29530/K32JB

7. Abstract

The purpose of the accident safety analysis is to identify and analyze a range of credible events, their cause and consequences, and to provide technical justification for the conclusion that Uranium billets, fuel assemblies, uranium scrap, and chips and fines drums can be safely stored in the 300 Area N Reactor Fuel Fabrication and Storage Facility, the contaminated equipment, High-Efficiency Air Particulate filters, ductwork, stacks, sewers and sumps can be cleaned (decontaminated) and/or removed, the new concretion process in the 304 Building will be able to operate, without undue risk to the public, employees, or the environment, and limited fuel handling and packaging associated with removal of stored uranium is acceptable.

The Accident Safety Analysis also provides a basis for the Facility TSRs which assure that the risk is acceptable.

8. PURPOSE AND USE OF DOCUMENT - This document was prepared for use within the U.S. Department of Energy and its contractors. It is to be used only to perform, direct, or integrate work under U.S. Department of Energy contracts. This document is not approved for public release until reviewed.

PATENT STATUS - This document copy, since it is transmitted in advance of patent clearance, is made available in confidence solely for use in performance of work under contracts with the U.S. Department of Energy. This document is not to be published nor its contents otherwise disseminated or used for purposes other than specified above before patent approval for such release or use has been secured, upon request, from the Patent Counsel, U.S. Department of Energy Field Office, Richland, WA.

DISCLAIMER - This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party's use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

9. Impact Level 2 S

10. RELEASE STAMP

OFFICIAL RELEASE
BY WHC
DATE JAN 24 1994
Station 21

MASTER

# ACCIDENT SAFETY ANALYSIS FOR 300 AREA N REACTOR FUEL FABRICATION AND STORAGE FACILITY

WESTINGHOUSE HANFORD COMPANY

JANUARY 1994

For the U.S. Department of Energy
Contract DE-AC06-87RL10930

This page intentionally left blank.

## CONTENTS

This page intentionally left blank.

## LIST OF FIGURES



## LIST OF TABLES

This page intentionally left blank.

# ACCIDENT SAFETY ANALYSIS FOR 300 AREA N REACTOR FUEL FABRICATION AND STORAGE FACILITY

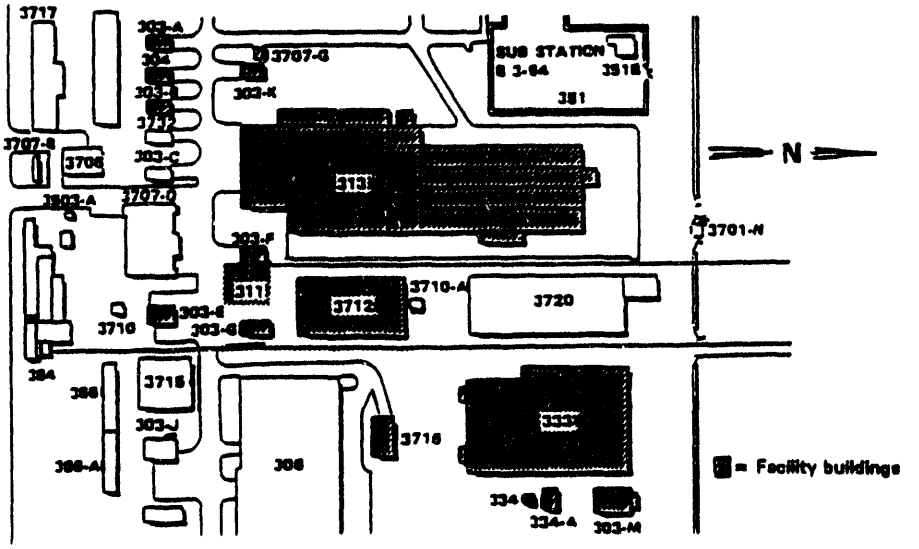
## 1.0 INTRODUCTION

An accident safety analysis was performed for the 300 Area N Reactor Fuel Fabrication and Storage Facility (Facility) to establish a technical justification for the Interim Safety Basis (ISB), (Brehm and Deobald 1994) conclusion that the Facility does not represent an undue risk to the public, employees, or the environment. In addition, the analysis provides a basis for the Facility Technical Safety Requirements (TSR), (Besser 1994). This report describes the manner in which the analysis was performed in accordance with WHC-CM-4-46, *Nonreactor Facility Safety Analyses Manual*, 7.0, "Risk", and WHC-CM-6-32, *Safety Analysis and Engineering Work Procedures*, WP-5.5, "Final Safety Analysis Reports," and describes the rationale upon which it was concluded that the cleanup activity, fuel handling and packaging, and fuel storage Facility are within acceptable risk guidelines.

## 1.1 BACKGROUND

The Facility, consisting of fuel fabrication buildings, laboratories, a concretion facility (mixing uranium fines and sludge with masonry cement), uranium and Zircaloy-2 fines incinerator, uranium Special Nuclear Material (SNM) storage buildings, and offices is located in the northeast corner of the 300 Area on the Hanford Site. A Facility layout is shown in Figure 1.1-1. Fuel fabrication and incinerator operations have been completely shutdown. The Facility is now used to store uranium billets, assembled and partially assembled fuel elements and scrap, and is also used for office space, see Table 1.1-1.

Figure 1.1-1. Facility Layout.



The Facility is currently undergoing transition activities required for permanent closure. The planning process for these activities has been documented in the Facility shutdown plan (Gimera 1992).

Table 1.1-1. Buildings Used for Special Nuclear Material Storage.

| BUILDING | FUNCTION | MTU[a] |
|---|---|---|
| 303-A | Green fuel[b] storage. (Fuel assemblies which were loaded into N Reactor but not irradiated) | 122 |
| 303-E | | 52 |
| 303-B | Uranium billet storage. | 289 |
| 303-G | | 250 |
| 303-K/ 3707-G North Room Pad | Drums of contaminated oil | <1 |
| | Chips & fines drums 113 L (30 gal) stored 0.61 m (2 ft) apart | |
| | Drums of mixed waste degreaser solvents. | |
| 313 | The SNM in 313 will be removed in the near future. | 257 |
| 3712 | Storage of uranium billets, clad fuel assemblies, clad fuel elements (scrap), beryllium braze rings, miscellaneous small parts.[c] | 1122 |
| 3716 | Unfinished fuel elements with plastic caps, wooden boxes, rags in cardboard boxes. | 137 |
| | TOTAL | 2,229 |

a. Metric Tons Uranium (inventory at the start of the analysis).
b. Green Fuel (unirradiated, potentially surface contaminated with activation and fission products).
c. Subsequent to this analysis, the beryllium braze rings and miscellaneous small parts have been removed.

### 1.1.1 Purpose

The purpose of the accident safety analysis is to identify and analyze a range of credible events, their cause and consequences, and to provide technical justification for the conclusion that:

- Uranium billets, fuel assemblies, uranium scrap, and chips and fines drums can be safely stored in the 300 Area N Reactor Fuel Fabrication and Storage Facility,

- The contaminated equipment, High-Efficiency Air Particulate (HEPA) filters, ductwork, stacks, sewers and sumps can be cleaned (decontaminated) and/or removed,

- The new concretion process in the 304 Building will be able to operate, without undue risk to the public, employees, or the environment, and

- Limited fuel handling and packaging associated with removal of stored uranium is acceptable.

The Accident Safety Analysis also provides a basis for the Facility TSRs which assure that the risk is acceptable.

### 1.1.2 Scope

This accident safety analysis addresses the consequences of a range of potential accidents connected with the storage and limited handling of uranium (billets, fuel assemblies, and scrap), the cleanup of the shutdown buildings, and the operation of the new concretion process (in the 304 Building), for the buildings listed in Table 1.1-1.

### 1.1.3 Methodology

A hazard analysis (Johnson and Brehm 1994), was initiated in 1992 by a team effort walk-through of the Facility to identify energy, radiological, toxicological, and other sources, and hazards. Members of the team included representatives from Operations, Restoration and Remediation Safety Analysis, Safety Technical Support, Risk Assessment Technology, and Facility Operations Safety Support. For each area of the Facility the following were identified: Hazards/Energy Sources, Potential Accident Sequences, Potential Target/Consequences, and Mitigating Barriers. This information was consolidated into a general hazard analysis covering natural events and hazards common to all buildings, and a specific hazard analysis for each building covering the hazards unique to that building. The accidents listed were then assigned severity (consequences) and probability categories for mitigated and unmitigated accidents. The hazard analysis serves as a checklist for the analyst while analyzing accidents and preparing the ISB.

The hazard class for the Facility, which was determined to be a Nuclear Facility with a Moderate Hazard Class rating (Huang 1993), established the review and authorization level of the safety analysis and provides a basis for applying a graded approach to the level of analysis and documentation. The hazard classification was based on unmitigated consequences, with no credit taken for administrative controls or protection systems planned or in place. The objective of the hazard classification is to assure that the safety analysis, review, and authorization levels applied are commensurate with the hazards potential.

Based on the hazard analysis (Johnson and Brehm 1994), a range of principal accident scenarios was selected for further review. Radiological and toxicological consequences of events involving the release of fuel

materials were based on the quantities and enrichments of uranium stored in the Facility at the time the hazard class was prepared. Since the hazard class, the overall Facility inventory including that of the 3712 Building has decreased. Impending uranium transactions may realize further inventory reduction. The radiological Curie content for each of the $^{235}U$ enrichments in the Facility inventory, 0.71 wt% (natural) , 0.95 wt%, and 1.25 wt%, were calculated. Curie (Ci) content for each enrichment for 1 metric ton of uranium (MTU) was approximately 1 Ci. A Radiological Consequence Code (GENII Analysis computer code) was used to calculate the dose consequences for a unit release of 1 MTU. The unit release of 1 MTU was then utilized for the calculation of dose consequences for the release associated with the various accident scenarios that were studied.

To develop the uranium isotopic composition for the Facility GENII analysis, a sensitivity assessment was made using a GENII analysis for a low-level radioactive solid waste burial ground, that was available at the time, to calculate the roentgen equivalent man (rem) consequences for each of the three enrichments (0.71, 0.95, and 1.25 wt% $^{235}U$). The rem release for 1 MTU of each of the three enrichments had less than 1% variation; therefore, instead of using all three enrichments for the Facility GENII analysis, the mixture for an enrichment of 1.25 wt% $^{235}U$ was chosen as the standard. This mixture; 0.009 wt% $^{234}U$ = 5.8 E-1 Ci, 1.25 wt% $^{235}U$ = 2.7 E-2 Ci, 0.069 wt% $^{236}U$ = 4.7 E-2 Ci, and 98.67 wt% $^{238}U$ = 3.5 E-1 Ci was used by the Radiological Safety Analysis group for the GENII analysis specific to the Facility, Appendix A.

The onsite distance was chosen as 100 m, the minimum distance that GENII analysis code can examine, although there are buildings within that limit. Two offsite locations were evaluated in the GENII analysis, the adjacent Columbia River edge (490 m) and a farm (1104 m) immediately across the river. This provided the dose consequences to a fisherman on the adjacent river bank and at the farm directly across the river. The $^{99}Tc$ in the fuel was of concern, due to its tendency to build up in uranium incinerator filters and its surface dose rate. For the effects of $^{99}Tc$ in the uranium, a second GENII analysis was made adding 10 ppm = 10 g/MTU $^{99}Tc$ to the mixture. The rem consequences changed only in the ingestion doses which are of interest only in developing the hazard classification. The final effective dose equivalents (EDE) in rem and the organ dose in rem for a unit release of 1 MTU + 10 g $^{99}Tc$ of the uranium mixture are given in Table 1.1.3-1. The X/Q values (calculated dispersion factors) are those that apply to the 300 Area, in general, and are also valid for the toxic releases.

A review of the literature on combustion of uranium and Zircaloy-2 is presented in Appendix B. Other supporting work included criticality safety studies (Schwinkendorf 1993), which addressed normal accident and upset conditions, including fire, bringing together multiple safe masses, mis-stacking, and accidental interspersed moderation. Also, a fire criticality probability analysis (Kelly 1994) was prepared, which addressed the probability of random and seismic induced fires without mitigation in conjunction with fuel storage box mis-stacking errors. Fire loading studies (Myott 1993) were performed to establish factors contributing to potential fire conditions.

Table 1.1.3-1. GENII Analysis, 1 MTU Unit Release Dose Consequences. (Appendix A)

| RECEPTOR | DISPERSION X/Q, $s/m^3$ | DOSE TYPE | EFFECTIVE DOSE EQUIVALENT (EDE) IN REM | ORGAN DOSE (REM) | LIMITING ORGAN |
|---|---|---|---|---|---|
| Onsite (100 m East), Nearest Occupied Building. | 3.4 E-2 | Inhalation | 1.5 E+3 | 1.2 E+4 | (Lung) |
| | | Submersion | 1.9 E-5 | 1.9 E-5 | |
| | | TOTAL | 1.5 E+3 | 1.2 E+4 | (Lung) |
| Site Boundary (490 m East) Fisherman at River's Edge. | 2.3 E-3 | Inhalation | 9.6 E+1 | 8.0 E+2 | (Lung) |
| | | Submersion | 1.3 E-6 | 1.3 E-6 | |
| | | TOTAL | 9.6 E+1 | 8.0 E+2 | (Lung) |
| Agricultural Area East of River (Autumn). (1104 m East) | 5.5 E-4 | Inhalation | 2.3 E+1 | 1.9 E+2 | (Lung) |
| | | | | 1.2 E-2 | |
| | | Submersion | 3.1 E-7 | 3.1 E-7 | |
| | | Ingestion | 5.0 | 1.0 E-1 | |
| | | | | 7.2 E+1 | (Bone surf) |
| | | Ground Shine | 3.3 E-3 | 3.3 E-3 | |
| | | TOTAL (Lung) | 2.8 E+1 | 1.9 E+2 | (Lung) |
| Agricultural Area East of River (Winter). (1104 m East) | 5.5 E-4 | Inhalation | 2.3 E+1 | 1.9 E+2 | (Lung) |
| | | | | 1.2 E-2 | |
| | | Submersion | 3.1 E-7 | 3.1 E-7 | |
| | | Ingestion | 2.7 E-2 | 1.4 E-3 | |
| | | | | 2.3 E-1 | (Bone surf) |
| | | Ground Shine | 3.3 E-3 | 3.3 E-3 | |
| | | TOTAL (Lung) | 2.3 E+1 | 1.9 E+2 | (Lung) |

NOTE: For a release of 1 MTU respirable mixture of 0.009 wt% $^{234}U$, 1.25 wt% $^{235}U$, 0.069 wt% $^{236}U$, 98.67 wt% $^{238}U$ and 10 ppm $^{99}Tc$ where 1.17 Ci = 1 MTU +10 ppm $^{99}Tc$ (GENII analysis).

### 1.1.4 Conclusions

The storage of uranium in the Facility, and the cleanup and transition activities required for permanent closure are well within the risk acceptance guidelines identified in WHC-CM-4-46. Radiological dose and toxicological consequences for the maximum credible event are less than Guidelines Consequences that would require Engineered Safety Features. There are no Safety Class 1 or 2 systems identified as described in WHC-CM-1-3, *Management Requirements and Procedures*, MRP 5.46, "Safety Classification of Systems, Components, and Structures," see Table 3.2-1. The accident safety analysis and the administrative controls that are a bases for these conclusions are described in Sections 2.0, 3.0, and 4.0.

## 2.0 HAZARDS IDENTIFICATION

The Facility no longer manufactures N Reactor fuel assemblies and is in transition to permanent closure. Current activities are centered around:

1. Storage of uranium Special Nuclear Materials (SNM),
2. Cleanup of facilities, and equipment,
3. Removal of equipment that can't be cleaned up,
4. Concretion plant operation to stabilize chips and fines now stored or to be recovered during the cleanup process, and
5. Potential uranium handling and packaging.

The SNM consists of uranium billets, clad reactor fuel assemblies, green fuel assemblies (loaded and removed from N Reactor without irradiation), clad fuel element scrap, unfinished fuel elements (clad but without end caps, or not completely assembled), and chips and fines from the cleanup process.

### 2.1 SUMMARY

The review of the hazard analysis (Johnson and Brehm 1994), identified several abnormal operations or events in which localized spills and releases could occur. These incidents include; handling fuel materials, cleaning activities, and fires involving small quantities of pyrophoric materials. These releases were found to be minimal and confined either within the buildings or immediately adjacent to the building. The accidents used in this analysis were identified on the basis of more serious injury or widespread dispersal of radiological and toxicological substance. Fires involving various forms of the stored fuel materials are the major events considered for further analysis in this study.

### 2.2 ABNORMAL OPERATIONS/EVENTS

The hazard analysis (Johnson and Brehm 1994), lists the possible hazards by building, and associated probabilities and severities (consequences) of events. Those considered abnormal operations or events which take place in buildings or which could have minor releases not affecting onsite or offsite personnel are listed in Table 2.2-1 and condensed into groups in Table 2.2-3.

Table 2.2-1. Abnormal Operations/Events. (Minor Radiological/ Toxicological Release Could Occur Locally, Contamination Could Be Spread, Personal Injury Could Occur)

| ABNORMAL OPERATIONS/EVENTS | RELEASE OR CONTAMINATION SPREAD | PERSONAL INJURY |
|---|---|---|
| 1. Clad fuel assembly dropped while handling. | No | Yes |
| 2. Uranium billet dropped during handling. | Yes | Yes |
| 3. Contamination spread while trying to remove or stabilize smearable contamination. | Yes | No |
| 4. Spill during sampling and/or characterization. | Yes | No |
| 5. Release of uranium, fission and/or activation products material due to accidental sprinkler trip. | Yes | No |
| 6. Spill during removal of suspected radioactive or toxic residue from drains, sumps and sewers. | Yes | Yes |
| 7. Spill and/or fire during removal of radioactive or toxic residue from equipment. | Yes | Yes |
| 8. HEPA filter rupture and/or fire during sampling or removal. | Yes | Yes |
| 9. Airborne release and/or fire while sampling or removing residual material from process exhaust ductwork. | Yes | Yes |
| 10. Fire during removal of residual uranium and Zircaloy-2 chips and fines. | Yes | Yes |

## 2.3 ACCIDENTS

The principal types of accidents identified in the hazard analysis (Johnson and Brehm 1994), are listed in Table 2.2-2. Those with low probability but serious consequences and those considered to have a fairly high probability were considered. Occurrences with a possibility of serious personal injury are listed even though they may have a capability for only a very low release or none at all.

In the first group, dropping a compressed gas bottle is listed mainly for its potential of serious injury and/or as an ignition source for a fire. The three natural occurring incidents all could start a fire or potentially increase the probability of a criticality; however, the amount of release would be due to a release of surface contamination and therefore minimal. The potential for criticality will be discussed in Section 4.2.4.

The collapse of the 313 Building south-end roof and the stack collapse are listed because of the possibility of serious personnel injury, but the radiological/toxicological release would be minimal.

Table 2.2-2. Accidents. (Local, Onsite, and Offsite Radiological and/or Toxicological Releases Could Occur)

| ACCIDENTS IDENTIFIED | RADIOLOGICAL/ TOXICOLOGICAL RELEASE | PERSONAL INJURY |
|---|---|---|
| 1. Bottle Gas Dropped - Regulator damaged or knocked off; fire during hydraulic oil removal. | No[a] | Yes |
| 2. Earthquake.[b] | No[a] | Yes |
| 3. Wind Storm.[b] | No[a] | Yes |
| 4. Flood.[b] | Yes | Yes |
| 5. Steam line rupture introduces steam into fuel storage area. (low-density moderator incident)[b] | Yes | Yes |
| 6. Collapse of the 313 Building south-end roof. | Yes | Yes |
| 7. Stack collapse during removal or because of deterioration and natural forces. | Yes | Yes |
| 8. Concreted chips and fines drum lid comes off exposing chips and fines allowing fire to occur. | Yes | Yes |
| 9. Concreted chips and fines drum ruptures exposing chips and fines allowing fire to occur. | Yes | Yes |
| 10. Explosion in chips and fines drum due to hydrogen production. | Yes | Yes |
| 11. Chips and fines drum dries out or spills allowing fire to occur. | Yes | Yes |
| 12. Fire in fuel storage building.[b] | Yes | Yes |
| 13. Criticality.[b] | Yes | Yes |

a. This accidents/events alone will not cause a release; however it may initiate other events which will.
b. Criticality is treated as a group to simplify discussion, all items marked [b], will be considered as possibly contributing to increasing the potential for a criticality.

## 2.4 CONCLUSIONS

Table 2.2-3 identifies abnormal operations or events and accident groups selected from Tables 2.2-1 and 2.2-2 on the basis of their potential for local, onsite, and offsite radiological and/or toxicological releases. Fire is the most serious mechanism for dispersal of radiological or toxicological substances. These abnormal operations or events and accident groups are analyzed in Section 4.0.

Table 2.2-3. Abnormal Operations/Events and Accidents.

| ABNORMAL OPERATIONS/EVENTS | RELEASE OR CONTAMINATION SPREAD | PERSONAL INJURY |
|---|---|---|
| 1. Spill while removing toxic and/or radioactive residue from drains, sumps and/or sewers. | Yes | Yes |
| 2. Release during removal of residual uranium fines, or radioactive or toxic residue from equipment. | Yes | Yes |
| 3. HEPA filter rupture and/or fire during sampling or removal. | Yes | Yes |
| 4. Airborne release or fire while sampling or removing residual material from ductwork. | Yes | Yes |
| 5. Release of uranium, fission, and/or activation products. | Yes | No |
| **ACCIDENTS** | **RADIOLOGICAL/ TOXICOLOGICAL RELEASE** | **PERSONAL INJURY** |
| 1. Chips and fines drum fire. | Yes | Yes |
| 2. Fire in fuel storage building(s). | Yes | Yes |
| 3. Criticality. | Yes | Yes |

## 3.0 DETERMINATION OF HAZARD AND SAFETY CLASSES

### 3.1 HAZARD CLASS

The hazard class for the Facility was determined to be a Nuclear Facility with a Moderate Hazard Class rating (Huang 1993).

### 3.2 SAFETY CLASS

The Facility systems, structures, and components, were determined to functionally be Safety Class 3 since the maximum onsite radiological and toxicological releases; 3.9 rem EDE, Be - 1.35 $\mu g/m^3$, and U - 3.1 $mg/m^3$, shown in Table 3.2-1, below the Safety Class 1 and 2 limits.

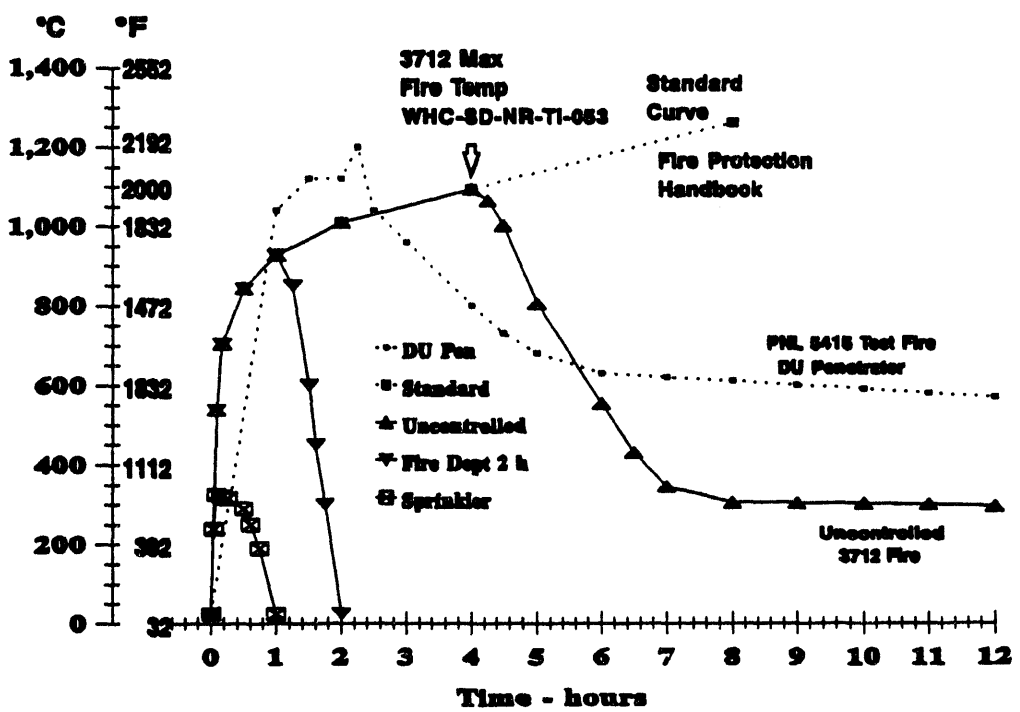
Safety Class 3 designation is based on the longest fire possible with the fire loading identified in the 3712 Building (Myott 1993). This maximum fire, with an 8 hour release, is depicted in Figure 3.2-1. For a longer or more intense fire to occur the fire loading would have to be increased.

Table 3.2-1. Safety Class Exposure Limits Comparison with Maximum Expected Facility Release.. (WHC-CM-1-3, MRP 5.46)

| SAFETY CLASS | RADIOLOGICAL CONSEQUENCES | | TOXICOLOGICAL CONCENTRATION | |
|---|---|---|---|---|
| | Onsite | Offsite | Onsite | Offsite |
| 1 | None. | > 500 mrem EDE | | > ERPG-2 |
| | | | | Be > 25 $\mu g/m^3$<br>U > 0.6 $mg/m^3$ |
| 2 | > 5 rem EDE | [a] | > ERPG-3 | ≤ ERPG-2 |
| | | | Be > 100 $\mu g/m^3$<br>U > 30 $mg/m^3$ | Be ≤ 25 $\mu g/m^3$<br>U ≤ 0.6 $mg/m^3$ |
| 3 | ≤ 5 rem EDE | [a] | ≤ ERPG-3 | |
| | | | Be ≤ 100 $\mu g/m^3$<br>U ≤ 30 $mg/m^3$ | |
| Maximum expected Facility release | 3.9 rem EDE | 250 mrem | Be - 1.35 $\mu g/m^3$<br>U - 3.1 $mg/m^3$ | Be - 0.091 $\mu g/m^3$<br>U - 0.21 $mg/m^3$ |

a. < 500 mrem EDE implied.

Figure 3.2-1. 3712 Building Estimated Fire Profiles.



# 4.0 ABNORMAL OPERATIONS/EVENTS AND ACCIDENT ANALYSIS

## 4.1 ABNORMAL OPERATIONS/EVENTS ANALYSIS

In abnormal operations or events (Table 2.2-1), material released will usually remain within a building where the cleanup activity is in progress. Those operations or events that allow a release outside the building will result in minimal release. Most of these involve removing suspected contamination which can consist of beryllium, uranium, or Zircaloy-2 along with dust and other residual material from equipment.

### 4.1.1 Spill While Removing Suspect Residue or Cleaning Sumps, Drains and Process Sewers

Many pieces of equipment, sumps, drains and sewers are contaminated with radioactive or toxic residue. The amount of radiological and/or toxicological release during cleanup will partly depend on how well procedures are written and followed. The hazard analysis (Johnson and Brehm 1994), lists all of the machining and cutting equipment for the 313 and 333 Buildings that may require cleanup.

Any cleaning solution that became radioactive or toxic and are released to the ground water that occurs is a RCRA/CERCLA violation. The use of water or other liquids while cleaning sumps, drains and process sewers can cause a radiological and/or toxicological release to the process sewer system, which eventually goes to groundwater and the river. The individual release would be minimal; however, accumulative releases could be significant. Vacuuming or other dry methods of cleaning would eliminate liquid releases; however, if static charges are generated and discharged, there is the potential for ignition of pyrophoric materials and associated airborne releases.

### 4.1.2 Release During Removal of Residual Fines, Radioactive, or Toxic Residues from Equipment

Machining and cutting processes result in fines which can be small enough to fall into seams or gaps that are part of the machine. Cleaning up these machines exposes the fines to air and can result in fire. The amount of fines in any one machine, as estimated by the Fuels Supply Facility Operations, will be no greater than in a high efficiency particulate air (HEPA) filter. See the following section for an analysis of the dose consequences. Fines removed are placed in a water-filled 113 L (30 gal) drum to a maximum depth of 10 cm (4 in.) The drum is closed with a vented lid. The fines drums are stored in the north room of the 303-K Building. A fines drum, fire would be localized and any resulting release would be minimal.

### 4.1.3 HEPA Filter Rupture During Removal

The HEPA filters for the Facility are inside buildings, except for the 303-K north room HEPA filter and this filter should only contain uranium oxides. The 303-M Building exhaust and the 333 Building beryllium exhaust

stacks to the outside have been blanked off. The fans for all exhausts to the outside are inoperable. A HEPA filter release has some potential for an onsite or offsite release. One method available that could reduce the hazard to all personnel while bagging out the HEPA filters is a temporary full enclosure with remote handling gloves (a greenhouse). If the HEPA filter is removed and bagged in a greenhouse the probability of a release or personnel contamination will be greatly reduced. Special Work Permits (SWPs) including respiratory equipment would be required.

The uranium billets and fuel assemblies contain 10 ppm of $^{99}$Tc which has been a problem in the past during uranium incinerator operations in the 303-M Building (shutdown), due to the oxides accumulating in filters. Where there is any indication that a HEPA filter might contain uranium it should be assumed that $^{99}$Tc is present. Since $^{99}$Tc oxides are volatile and highly soluble, with water, forming pertechnic acid, and other solvents, any procedures for removal of filters that might be contaminated with $^{99}$Tc should contain special precautions for personnel safety. The contamination rate to Facility workers due to the chemical properties of $^{99}$Tc has been of concern in the past; however, its contribution to onsite and offsite dose rates is negligible, the GENII analysis using $^{99}$Tc changed only in the ingestion dosage which are of interest only in developing the hazard classification.

During removal of a HEPA filter, a rupture may occur, especially if the filter has been in service for a long time or is overloaded. The primary hazard from a ruptured HEPA filter is the possibility of a respirable release, either directly or as a result of a fire. The possibility of pyrophoric materials in the filter increases the fire hazard since spontaneous ignition of any uranium/Zircaloy-2 chips and fines can occur when the material is exposed to air.

The amount of material in a HEPA filter is estimated by Fuel Supply Facility Operations based on the filter size, as being less than 4.5 kg (10 lbs). Using a fractional release for moderately dispersible airborne material of 0.01 (WHC-CM-4-46, Section 4) and one of 8.9 E-8/s for pyrophoric materials (NRC 1988), two events have been reviewed; an airborne release, and a fire. With an onsite effective dose equivalent of 1.5 E+3 rem/MTU and 9.6 E+1 rem/MTU for a 490 m offsite location (from the GENII analysis, Table 1.1.3-1), and disregarding the confinement due to the building, airborne releases become:

4.5 kg x 1 MTU/1000 kg x 0.01 x 1.5 E+3 rem/1 MTU = 6.75 E-2 rem onsite, and

4.5 kg x 1 MTU/1000 kg x 0.01 x 9.6 E+1 rem/1 MTU = 4.3 E-4 rem offsite.

For an airborne release and assuming a 30 minute fire (estimated by Fuel Supply Facility Operations personnel), the airborne releases become:

4.5 kg x 1 MTU/1000 kg x 8.9 E-8/s x 60 s/min x 30 min x 1.5 E+3 rem/1 MTU = 1.1 E-3 rem onsite, and

4.5 kg x 1 MTU/1000 kg x 8.9 E-8/s x 60 s/min x 30 min x 9.6 E+1 rem/1 MTU = 6.9 E-5 rem offsite.

Calculations assume all the material is uranium, even though the material contains some Zircaloy-2.

### 4.1.4 Process Exhaust Ductwork Airborne Release

The process exhaust ductwork in the 313, 333, 304, and 303-M Buildings contains residue from manufacturing fuel assemblies, process laboratories, concretion, and uranium/Zircaloy-2 incineration. This residue could be released during sampling procedures and/or cleanup of the ductwork and baghouses. Use of a greenhouse or drapes would reduce the possible release to a minimum. Also, SWPs may be required.

The primary hazard of a release from the ducts is the respirable fraction released. The size of the release is dependent on the amount of material in the duct, which is unknown. It is unlikely the ducts would be able to support large amounts of uranium, due to the weight of the metal and oxide. For $^{99}Tc$ oxides, see above discussion on HEPA filters.

Cleaning procedures can increase the possibility of a release. Vacuuming the ducts with HEPA filtered vacuum cleaners would remove most of the loose material that could be released; however, if static electricity is generated and discharges during vacuuming the pyrophoric material could catch fire. Plated material is not apt to catch fire unless removed by scraping. For long ducts without access, sections may have to be removed to reach all loose material.

Ductwork releases could also result from fires external to the ductwork. The fabrication facilities and equipment are primarily constructed of non-combustible materials. However, there are combustible materials associated with the offices, stored materials, and hydraulic equipment and there is the potential for fires associated with electrical equipment and other maintenance activities. Fires could result in the removal of ductwork residues. Process equipment ductwork has been blanked-off to the outside.

### 4.1.5 Release of Uranium, Fission, and/or Activation Products

Buildings 303-A and 303-E are used for storage of green fuel, that is fuel assemblies that have been loaded and removed from N Reactor without irradiation. The outer assemble surface was scrubbed, rinsed well with water, and wiped dry during removal from N Reactor. The fuel was then covered with a plastic wrap that would provide some protection from release of contamination. These assemblies potentially have fission and activation product surface contamination which could be released by a sprinkler system trip, or flooding. This will spread a minimal amount of contamination on the surrounding surfaces. Other buildings having surface uranium contamination, if flooded, could release minimal levels of contamination to the ground or sewers.

## 4.2 ACCIDENT ANALYSIS

The accidents analyzed are listed in Table 2.2-3. Table 1.1-1 lists the buildings used for uranium SNM storage and the amount of SNM storage at the

time the hazard classification was performed. The SNM is stored in the form of uranium billets, clad fuel assemblies, clad fuel elements (scrap), unfinished fuel elements with plastic capped ends, chips and fines stored under water, and chips and fines imbedded in concrete. The chips and fines are pyrophoric and because they are designated as mixed-waste, the chips and fines drums are isolated from other SNM storage. The 303-K Building has 24 drums of chips and fines separated by a space of about 61 cm (2 ft) between drums. The 303-K Building is an active mixed-waste storage facility and subject to regulatory requirements for the storage of dangerous waste. The mixed waste is stored in U.S. Department of Transportation-specification drums. The drums containing chips and fines are regulated under a Part A Dangerous Waste Permit Application and a Resource Conservation and Recovery Act (RCRA) Closure Plan (DOE-RL 1991).

The SNM is stored in locked, unoccupied buildings except when the buildings are opened for the annual inventory, fire system surveillance, criticality inspections, the SNM is being moved to a different location, or the SNM is being packaged for shipment. Access to the SNM storage buildings is restricted. All fuel storage buildings have fire detection, alarm, and suppression systems (dry pipe sprinklers, except for the 303-K Building which does not have a suppression system, see Section 4.2.1). The buildings are unheated, although there are heaters for the dry pipe sprinkler standpipes. Buildings 3712 and 3716 have steam lines which are shut off. Building 3712 has an HVAC unit and process exhaust system which are inoperable. All the buildings have electric wiring and lights in the ceilings. Combustibles in the buildings consist of the wooden storage boxes and plastic wrappings for the uranium billets, fuel elements, and fuel assemblies. Smoking is prohibited in all buildings.

### 4.2.1 Chips and Fines Drum Fire

Uranium and Zircaloy-2 chips and fines are pyrophoric; therefore, anytime they are allowed to dry out, exposing them to the air, they can catch fire. Fires in chips and fines drums are much more likely than a fire in a fuel storage building. Past fires have been caused by allowing the chips and fines drums to dry out, and incorporation of excess chips and fines in the concrete mixture in conjunction with improper curing causing the concreted drums to split. During cleanup activities, any chips and fines located are designated as mixed waste, which is regulated by RCRA and must be stored accordingly. The chips and fines drums are stored in the 303-K Building which is an active mixed-waste storage facility. There are no billets or fuel assemblies stored in the 303-K Building; likewise, there are no chips and fines drums located in the billet or fuel-containing storage buildings. To do otherwise, in either case, would violate the regulatory storage requirements for those buildings. Due to the physical separation of these materials, it is incredible that a chips and fines drum fire will be a source of ignition for a billet or fuel assembly fire.

Hydrogen build-up in the water filled chips and fines drums has caused explosions at other sites. The water-filled chips and fines drums are covered by lids with HEPA filter vents which retard evaporation while reducing hydrogen buildup. Because the filters do not release all of the hydrogen, the drums are vented by removing the HEPA filter vent, on a monthly basis. The

basis for the monthly venting is, after years of storage the drums were checked and two of the drums had hydrogen buildup in the explosive range. Therefore, the monthly venting was initiated through a documented surveillance procedure; checking of the water level is a part of this same surveillance.

The chips and fines stored in the 303-K Building came from cleanup of residual material. Only 113 L (30 gal) drums were used for the cleanup materials. These drums average 104.5 kg (230 lb) of chips and fines. Assuming a one hour fire and using a fractional release (NRC 1988) of 8.9 E-8/s for pyrophoric material, and with an onsite effective dose equivalent of 1.5 E+3 rem/MTU and 9.6 E+1 rem/MTU for an offsite location (from the GENII analysis, Table 1.1.3-1), disregarding the confinement due to the building, the airborne dose consequences become:

104.5 kg x 1 MTU/1000 kg x 8.9 E-8/s x 60 s/min x 60 min x 1.5 E+3 rem/1 MTU = 0.05 rem onsite, and

104.5 kg x 1 MTU/1000 kg x 8.9 E-8/s x 60 s/min x 60 min x 9.6 E+1 rem/1 MTU = 3.2 E-3 rem offsite.

Although the chips and fines drums contain some Zircaloy-2, cleanup waste dirt and other non-nuclear materials, the airborne dose consequences was calculated as if the weight was entirely from uranium. The frequency of chips and fines drums fires is estimated to be 1E-03 and is based on Facility experience and precautionary measures that have been taken (protected storage, covered drums, and monthly surveillance of drum water levels).

The new concretion procedure will use masonry cement which contracts on curing and gives off about one-half the heat of Portland Type II cement during curing. This makes it less likely for continued chips and fines oxidation and associated drum rupture which may cause a fire to start. The amount of chips and fines placed in the drum being concreted, 14 kg (31 lb), makes the consequences very low if fire should start, due to the lower fractional release and weight. Assuming a one hour burn and a fractional release (FR) of 1.45 E-4 (Huang 1993) see Table 4.2-2, the airborne releases become:

14 kg x 1 MTU/1000 kg x 1.45 E-4 x 1.5 E+3 rem/1 MTU = 3 E-3 rem onsite, and

14 kg x 1 MTU/1000 kg x 1.45 E-4 x 9.6 E+1 rem/1 MTU = 0.2 E-3 rem offsite.

Spills may occur during the concretion procedure when drums are rotated or when fines are being emptied into the drum to be concreted but these would be contained within the building by the operators present.

The frequency of concreted drums fires is estimated to be 1E-04 and is based on Facility experience and precautionary measures that have been taken (reduction of chips and fines content, use of masonry cement, and controlled curing).

### 4.2.2 Fire in Fuel Storage Building 3712

The 3712 Finished Fuel and Billet Storage Building was analyzed because it contains the largest amount of combustible material (wooden fuel storage boxes) and the most uranium fuel and therefore, the potential for the highest onsite and offsite dose consequences." Building 3712 is a one story steel frame structure, 27.4 m by 32.9 m (90 ft by 108 ft), with metal panel siding and roof and a concrete floor and foundation. It is equipped with an automatic fire alarm and sprinkler (dry) system with freeze protection for the stand pipe and valve of the dry type sprinkler system. Figure 3.2-1 compares three possible scenarios for a fire in the 3712 Building. The standard time-temperature curve for a structure fire is identified in the fire loading (Myott 1993). The estimated maximum temperature a fire in 3712 Building could reach is analyzed to be 1093.3 °C (2000 °F) and after approximately 4 hours the free combustibles would be consumed.

In Appendix B, a comparison was made between a test fire involving depleted uranium penetrators and the burning of the billets and fuel elements in the 3712 Building. Uranium will oxidize while there is fuel present but does not support combustion when the free combustibles are removed. From the report (Hooker, et al., 1983), "The penetrators would not be expected to ignite until the temperature greatly exceeded 700 °C (1292 °F), if they can ignite at all." The fire loading for the penetrators was calculated to be 6 E+4 Cal wood/g penetrator (1.1 E+5 BTU wood/lb penetrator), while the fire loading for 3712 Building is 4.6 E+2 Cal wood/g uranium (830 BTU wood/lb uranium). Therefore, the uranium billets and fuel elements storage boxes consume only 0.77% of the free combustible supply per unit weight as compared to the penetrator test fire indicating much lower oxidation temperatures and a shorter cool down time for the 3712 Building.

The cooldown rate depicted in Figure 3.2-1 shows the penetrators cooling down slower than the billets and fuel elements. The fire loading for the two cases would account for part of the difference; however, the configuration of the burn would also be a factor. The green railroad ties, approximately 15 cm x 20 cm x 244 cm (6 in. x 8 in. x 8 ft), used in the penetrator fires were piled under, on top of, and around the 12 missiles and their iron framework sitting in a 9 m x 9 m (30 ft x 30 ft) rimmed steel tray. When the railroad tie fire collapsed after about 4 hours, the embers continued to emit heat and about 24 hours after the ignition of the pile the temperature was still about 350 °C (Hooker, et al., 1983), the material covered about 3 m x 3 m (10 ft x 10 ft).

With the amount of scattering of the billets, fuel elements, and boxes, as the boxes disintegrate in the fire, most of the 903 $m^2$ (9720 $ft^2$) floor area of the 3712 Building could be covered. The windows of the 3712 Building would blowout soon after the fire started, allowing free movement of air into the building and the escape of heat. The heat retention of the uranium and the boxes, even if the 3712 Building roof collapsed over the top of them, could not be maintain as high as the temperatures associated with the railroad ties in the penetrator fire (based on the relative fire loading, see first paragraph Section 4.2.2).

During the cooldown period there can be oxidation while the uranium metal remains above approximately 300 °C (572 °F). The amount of release during the

period the uranium is above approximately 300 °C (572 °F) will vary with temperature, however this variation was not considered in the release calculations. The release time of eight hours covers the time the billets and fuel elements are above 300 °C (572 °F).

To obtain a source term for a billet and fuel element fire, the emission duration was used to determine the amount of the total material consumed. Using empirical equations derived for uranium test specimens (Hilliard 1958) as shown in Appendix B, and the 48 hour duration of the free burn test of depleted uranium penetrators (Mishma, et al., 1985), a time period of 100 hours (Appendix B) was estimated for full combustion of uranium billets and fuel elements. For example, assuming a 2 hour fire and 1122 Metric Tons Uranium (MTU) exposed to the fire, the oxidized uranium is 1122 MTU x 2 h/100 h = 22.4 MTU.

The fractional release, 1.45 E-4 (Huang 1993), of the oxidized uranium used is a conservative choice based on studies of plutonium. A respirable fraction of 0.2 was used (Mishima, et al., 1985) which gives a combined fractional release of 2.9 E-5.

The frequency of a random fire burning unabated in the 3712 Building, the 8 hour fire, is 1.6 E-7 (Table 4.2.2-1) which puts it into the incredible class (WHC-CM-4-46). If a random fire does occur and the sprinkler system or Hanford Fire Department Station No. 93, 0.4 km (0.25 mi) away, or Hanford Fire Department Station 94, 9.6 km (6 mi) away responds (event probability of between 1.6 E-3 and 1.6 E-5) fire suppression should start relatively early. Assuming a 2 hour fire and based on a 2.9 E-5 fractional release, the onsite and offsite dose airborne dose consequences (Table 4.2.2-2), for the random fire is:

1122 MTU x 2 h/100 h x 2.9 E-5 x 1.5 E+3 rem/1 MTU = 0.98 rem onsite, and

1122 MTU x 2 h/100 h x 2.9 E-5 x 9.6 E+1 rem/1 MTU = 0.062 rem offsite

The amount of uranium involved, when the sprinkler system operates as planned or the fire departments respond, will depend on the location of the initial fire, however, the actual release should be minimal.

The seismic event with the fire department not responding has a frequency of 5.55 E-8. This is also in the incredible class (WHC-CM-4-46). For this Beyond Design Basis Event (BDBE), assuming a four hour fire [the Fire Loading Analysis (Myott 1993) shows there is only sufficient free combustibles for a 4 hour fire] and a 8 hour uranium oxidation time, and based on a 2.9 E-5 fractional release (Table 4.2.2-2), the onsite and offsite dose airborne dose consequences for the BDBE seismic fire is:

1122 MTU x 8 h/100 h x 2.9 E-5 x 1.5 E+3 rem/MTU = 3.9 rem onsite, and

1122 MTU x 8 h/100 h x 2.9 E-5 x 9.6 E+1 rem/MTU = 0.25 rem offsite

While the actual release will occur only while the uranium metal temperature is above approximately 300 °C (572 °F) and will vary with temperature, for these calculations the release time has been taken as the

time from ignition to the time when the uranium temperature has dropped below 300° C (572 °F).

The uncontrolled fire, Figure 3.2-1, shows the temperature dropping to about 300 °C (572 °F) after eight hours, at which time the release would stop. This gives an onsite release of 3.9 rem (see Table 3.2-1). With the combustibles consumed after four hours and the scattering of uranium and ashes, the temperature drop will be fairly rapid. For oxidation to last longer than eight hours, additional combustibles or flammable material must be introduced. The lower limits for Safety Class 1 and Safety Class 2 have not been violated (see Tables 3.2-1, 4.2.2-2, and 4.2.2-3), therefore, there is no requirement for either Safety Class 1 or 2 equipment. The analysis does not take credit for the uranium oxidation protection that would be provided by the Zircaloy-2 cladding on the finished fuel assemblies.

Table 4.2.2-1. Fuel Supply Facilities Fire Analysis Event Frequency. (Kelly 1994)

| RANDOM FIRE | | SEISMIC EVENT CAUSED FIRE | |
|---|---|---|---|
| EVENT | FREQUENCY | EVENT | FREQUENCY |
| Fire in the 3712 Building | 0.16 | Seismic event | 1.11 E-3 |
| | | Seismic event causes fire in 3712 | 1.0 E-2 |
| Sprinkler System Fails | 1.00 E-2 | Reaches fuel | 0.5 |
| Fire Department fails to respond. | 1.00 E-4 | Seismic event fails the non Safety Class 1 fire system and/or associated supply system. | 1.0 |
| | | Fire Department fails to respond | 1.0 E-2 |
| All events | 1.6 E-7 | All Events | 5.55 E-8 |

The radiological releases for the fires discussed are shown in Table 4.2.2-2. The toxic release concentrations are shown in Table 4.2.2-3. Plots of these results in terms of radiological, uranium, and beryllium risk, and the frequency of the fires are shown in the set of curves in Figures 4.2.2-1, 4.2.2-2, and 4.2.2-3, which give rem releases, and toxic concentrations for uranium and beryllium respectively. In all cases, the release used is based on the fire length given, although the release will not start until the temperature exceeds 300 °C (572 °F) and will stop when the temperature drops below 300 °C
(572 °F).

## Table 4.2.2-2 Facility Accident Radiological Dose Consequences for a Range of Credible and Incredible Fires.

| FIRE DURATION | EXPOSED MTU[a] | OXIDIZED MTU[a] | RELEASED MTU[a] | ONSITE 100 M EDE[c] REM | | OFFSITE RIVERS EDGE 490 M EDE[c] REM | | OFFSITE FARM ACROSS RIVER IPR AUTUMN 1140 M EDE[c] REM | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | Inh[c] (Total) | Sub[c] | Inh[c] (Total) | Sub[c] | Inh[c] | Sub[c] | Ing[c] | Total |
| | | | 1[b] | 1.5 E+3 | 1.9 E-5 | 9.6 E+1 | 1.3 E-6 | 2.3 E+1 | 3.1 E-7 | 5.0 | 2.8 E+1 |
| Chips and Fines Drums FR = 8.9 E-08/s x 60 min x 60 sec = 3.2 E-4 (from NUREG 1320 Table 4.2 Pyrophoric metals) | | | | | | | | | | | |
| 1 hour | 0.105 | 0.105 | 3.4 E-5 | 0.051 | 6.4 E-10 | 3.22 E-3 | 4.4 E-11 | 7.7 E-4 | 1.1 E-11 | 1.7 E-4 | 9.4 E-4 |
| Concreted drum FR = 1.45 E-4 (Huang 1993)(no credit taken for respirable portion.) | | | | | | | | | | | |
| 1 hour | 1.4 E-2 | 1.4 E-2 | 2.1 E-6 | 3.1 E-3 | 3.9 E-11 | 2.0 E-4 | 2.7 E-12 | 4.8 E-5 | 6.4 E-13 | 1.0 E-5 | 5.80 E-5 |
| 3712 Bldg Fractional Release FR = 1.45 E-4 x 0.2 = 2.9 E-5 (Huang 1993)(0.2 is the respirable portion see Table 9-2) | | | | | | | | | | | |
| 2 hour | 1122 | 22.4 | 6.5 E-4 | 0.98[d] | 1.24 E-8 | 0.062 | 8.46 E-10 | 1.5 E-2 | 2.2 E-10 | 3.2 E-3 | 1.8 E-2 |
| 8 hour | 1122 | 89.8 | 2.6 E-3 | 3.9 | 4.95 E-8 | 2.5 E-1 | 3.4 E-9 | 0.0060 | 8.1 E-10 | 1.3 E-2 | 7.3 E-2 |

a. MTU - Metric Ton Uranium
b. GENII analyses were based on a unit release of 1 Ci, where 1 MTU = 1 Ci for all cases. The isotopic composition of the uranium was as follows: 0.009 wt% $^{234}U$, 1.25 wt% $^{235}U$, 0.069 wt% $^{236}U$, 98.67 wt% $^{238}U$, and 10 ppm $^{99}Tc$.
c. Inh - Inhalation, Sub - Submersion, Ing - Ingestion, EDE - Effective Dose Equivalent
d. Example calculation: 1122 exposed MTU x 2 hr fire/100 hr fire for total oxidation x 2.9 E-5 respirable MTU released/MTU oxidized x 1.5 E+3 REM/respirable MTU released = 0.98 REM

Note: Accident analyses are limited to one building at a time based on the fire hazard analysis (Myott 1993) that shows that the separation between buildings is adequate to prevent the propagation of fires between buildings.

Table 4.2.2-3. Toxic Risk Comparison with Emergency Response Planning Guidelines. (ERPG)

| EMERGENCY RESPONSE GUIDELINES (ERPG)/ FACILITY RESULTS[a] | ONSITE (100 m) Moderate Criteria (ERPG-2 < X ≤ ERPG-3) | OFFSITE RIVERS EDGE (490 m) Moderate Criteria (ERPG-1 < X ≤ ERPG-2) |
|---|---|---|
| BERYLLIUM GUIDELINE ($\mu g/m^3$) | 25 < X ≤ 100 | 2[b] < to ≤ 25 |
| Facility (0.494 MTBe[c]) | 1.35 | 0.091 |
| URANIUM GUIDELINE ($mg/m^3$) | 0.6 < X ≤ 30 | 0.05[b] < X ≤ 0.6 |
| Facility (1122 MTU[c]) | 3.1 | 0.21 |

a. All release concentrations based on 100 hours for total oxidation of the uranium.
b. These are listed as "not appropriate" below the moderate criteria level. The OSHA PEL 8 hr. limit is shown for comparison.
c. MTBe - Metric tons beryllium, MTU - Metric tons uranium.

### 4.2.3 Fire Propagation Between Adjacent Fuel Storage Buildings

To the north of the 306 West Building (a Pacific Northwest Laboratories facility) and 9 m (30 ft) directly to the east of the 3716 Building is a 500 gal above-ground propane storage tank, see Figure 1.1-1. The tank has been inspected and appears to be within the standard Department of Transportation requirements for flammable gases. The tank has four inch steel legs and stands on a concrete pad. It is located on the side of the facility where traffic is infrequent; the only identification need for vehicle traffic would be for the purpose of filling the tank. It is anticipated that if the line from the tank was to break, the propane flow would be adequately restricted so that large amounts of gas would not collect to propagate a major fire or explosion. If the gas stream was ignited the flame would be more like a torch out of the tubing and there would be no potential for propagation to the 3716 Building due to distance separation.

In the extremely unlikely event that the random fire was a fire not associated with the propane tank, it is conceivable that the tank could over heat to the extent that the propane tank would rupture and a fire ball would occur. This in turn conceivably could result in a fire in the 3716 Building. This building has a fuel capacity of 250 MTU; therefore, the dose consequences for this event would be approximately one fourth that of the fire event for the 3712 Building analyzed in the previous section. The Fire Hazards Analysis (Myott 1994) does not show that there is a capability for propagation of fires between buildings.

Figure 4.2.2-1. Radiological Risk Acceptance Guidelines with Accident Dose Consequences in REM.

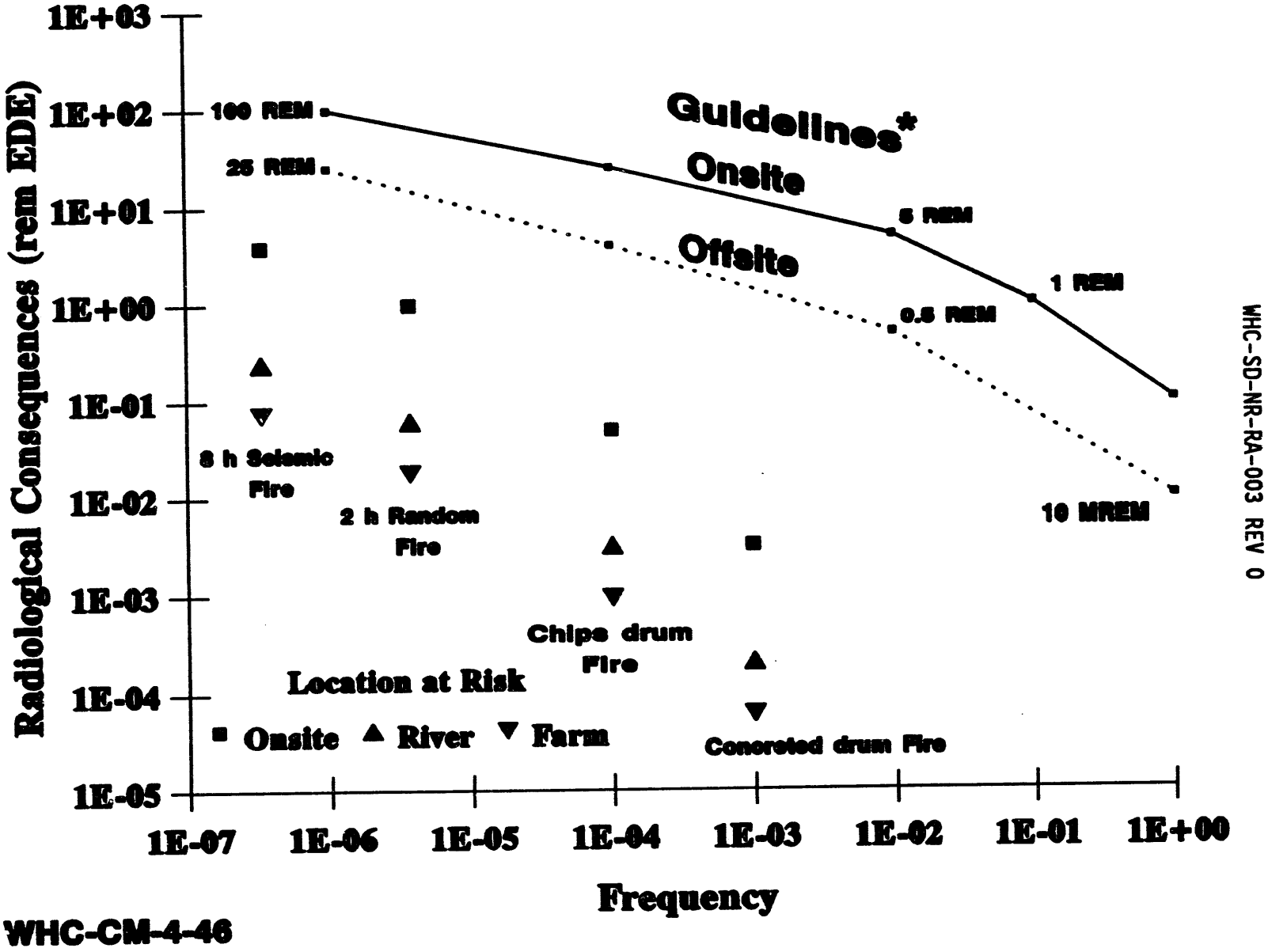
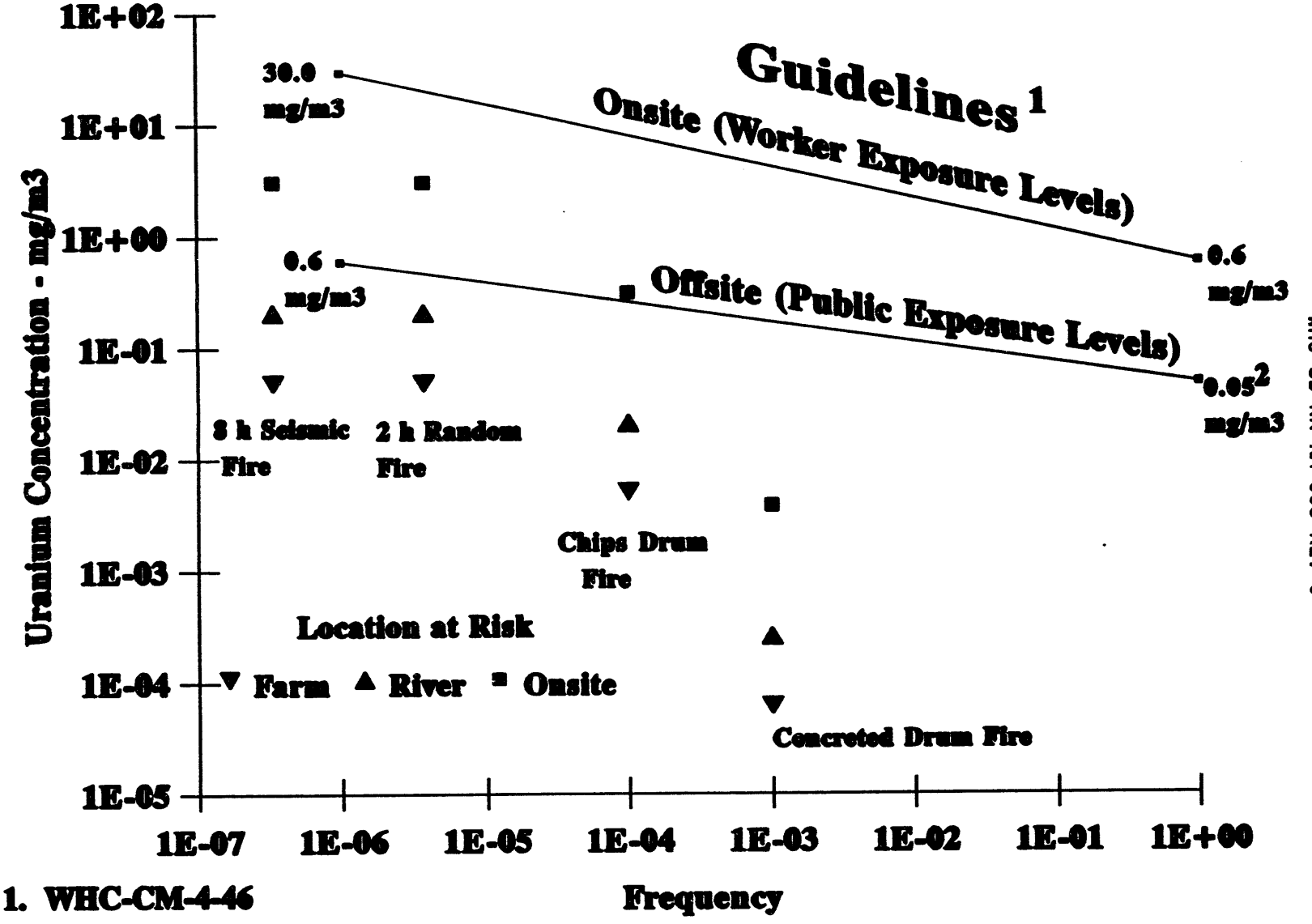
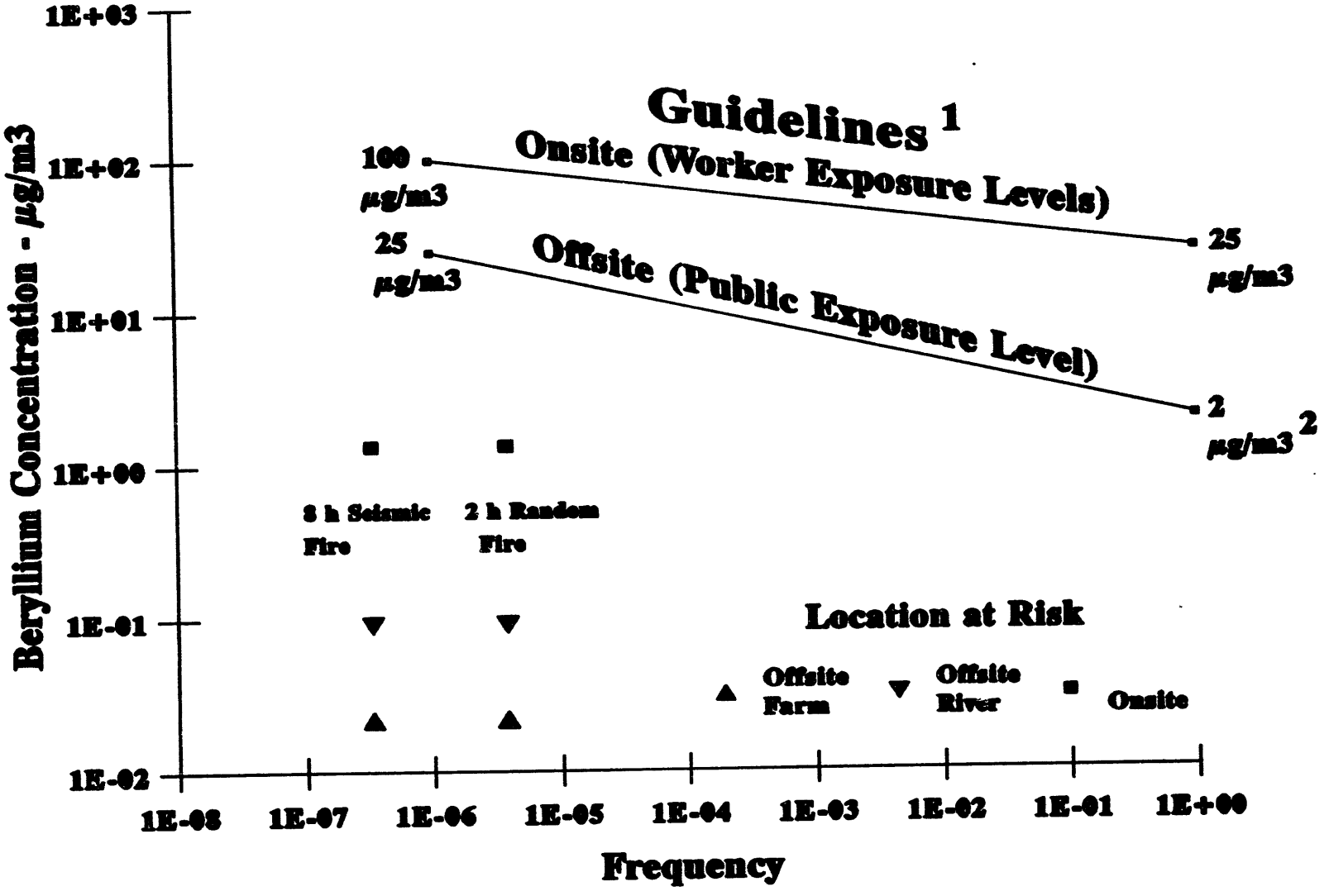


Figure 4.2.2-2. Uranium Risk Comparison Guidelines with Accident Release Concentrations.




1. WHC-CM-4-46
2. OSHA PEL (permissible exposure limit - 8 hour)

Figure 4.2.2-3. Beryllium Risk Comparison Guidelines with Accident Release Concentrations.




1. WHC-CM-4-46
2. OSHA PEL (Permissible exposure Limit - 8 hour)

There is also the extremely unlikely potential that the propane tank could be mechanically damaged, over filled, or overheated by means other than a fire with failure of the two pressure relief valves, such that the tank ruptures and a Boiling-Liquid Expanding-Vapor Explosion (BLEVE) occurs. The tank is located in an area where there is relatively little traffic, there are no overhead structures or cranes nearby, and there have been no known reports of BLEVEs associated with tanks of this relatively small size. In addition, because of the tank position with respect to the 3716 Building and the 3712 Building it is felt that the 3716 Building would shield the 3712 Building from the BLEVE. Considering the approximate 61 m (200 ft) separation between the tank and the 3712 Building, it is considered extremely unlikely for simultaneous fires to occur in the 3716 and 3712 Buildings.

None the less, if simultaneous unabated fires were to occur in the 3716 Building at its storage capacity of 250 MTU and the 3712 Building at its analyzed capacity of 1122 MTU, then the resulting onsite and offsite dose consequences could potentially be:

3.9 rem + 3.9 rem/1122 MTU X 250 MTU = 4.8 rem onsite, and

0.25 rem + 0.25 rem/1122 MTU X 250 MTU = 0.31 rem offsite

Safety Class 1 or 2 protective features are not required as dose consequences are less than the WHC-CM-4-46 guideline criteria of 5 rem onsite and 0.5 rem offsite.

### 4.2.4 Criticality

Criticality safety calculations (Schwinkendorf 1993) have been performed to confirm and update the safety limit values currently found in WHC-NR-4-4, *Nuclear Safety Specifications*. These new values were produced using more modern computer codes that comply with Software Quality Assurance (SQA) requirements. In addition, certain accident, or upset conditions were analyzed. These scenarios included fire, the bringing together of multiple safe masses into one neutronically coupled system, mis-stacking, and accidental inter-dispersed moderation.

In the event of a fire and partial oxidation of uranium metal, uranium oxide addition to the surrounding water only serves to reduce reactivity compared to the case with pure water moderator. New limits as contained in the analysis are all based on the most conservative assumptions of optimum moderation by water and infinite reflection. In the event of total combustion and removal of the wooden boxes, the remaining uranium fuel would collapse into a slab substantially below the height required for $k_{eff}$ = 0.98 [The Criticality safety calculations (Schwinkendorf 1993) provides a bases for using $k_{eff}$ = 0.98 instead of the typical $k_{eff}$ = 0.95]. Even if this collapsed array was formed from an incorrectly stacked array (three high) the slab would still be at the criterion $k_{eff}$ = 0.98 (as long as one were to use bare slab heights). If the array boxes were only partially mis-stacked (e.g., one out of two, or four out of five columns), the collapsed array would be within the $k_{eff}$ = 0.98 criterion (a full third layer is just at this limit: $k_{eff}$ = 0.982 for a collapsed optimally-spaced bare slab).

Storage box arrays of MKIA assemblies have been analyzed. The MKIA fuel assemblies are the most reactive fuel because they contain 1.25 wt% enriched outer elements, and are therefore a bounding case. Under normal conditions, calculated $k_{eff}$ is substantially subcritical. Even under water flooding conditions (either full density or interspersed moderation of optimum density), storage array $k_{eff}$ is less than 0.90, even for arrays that are infinite in length and width, and stacked three boxes high. The corresponding $k_{eff}$ for two boxes high (the maximum allowed by criticality safety specifications) never exceeds 0.80. Boxes stacked to five high, with out interspersed moderation, reached a $k_{eff}$ of 0.955 and a $k_{eff}$ of 0.963 with $2\sigma$. Introduction of interspersed moderation would reduce the $k_{eff}$ to less that 0.95 because of the thermal versus absorption effects of water introduction in small versus large arrays.

Table 4.2.4-1. WHC-SD-NR-CSER-010 Table 1 - Safe Masses and Associated Dimensions.

| | Sphere | | Hemisphere | | Infinite Cylinder | | Infinite Slab | |
|---|---|---|---|---|---|---|---|---|
| | Mass(lb) | Dia. (in) | Mass (lb) | Dia (in) | Mass (lb/ft) | Dia (in) | Mass(lb/ft2) | Thick (in) |
| MKIV Fuel Assemblies (0.95 wt% uranium inner and outer elements) | | | | | | | | |
| Billets | ---------- | | ---------- | Unlimited | ---------- | | ---------- | |
| Assemblies | 5228 | 34.5 | 8575 | 51.3 | 1170 | 22.6 | 322 | 9.8 |
| Outer elements | 3605 | 33.2 | 5923 | 49.2 | 839 | 21.7 | 241 | 9.5 |
| Inner elements | 3084 | 30.6 | 5122 | 45.7 | 771 | 19.9 | 239 | 8.9 |
| Scrap | 2150 | 26.6 | 3603 | 42.8 | 567 | 19.4 | 180 | 9.1 |
| Solution | ---------- | | Unlimited | | ---------- | | ---------- | |
| MKIA Fuel Assemblies (0.95 wt% uranium inner element, 1.25 wt% uranium outer element) | | | | | | | | |
| Billets | 9859 | 35.9 | 15967 | 53.2 | 2134 | 23.6 | 584 | 10 |
| Assemblies | 1597 | 24.1 | 2716 | 36.4 | 486 | 15.9 | 187 | 6.8 |
| Outer elements | 932 | 22.1 | 1619 | 33.5 | 308 | 14.5 | 127 | 6.1 |
| Inner elements | 3176 | 30.7 | 5284 | 45.7 | 775 | 20.8 | 237 | 8.7 |
| Scrap | 586 | 20.1 | 1037 | 30.7 | 208 | 13.4 | 88 | 5.9 |
| Solution | 1503 | 28.1 | 2539 | 42.2 | 809 | 18.2 | 130 | 8.9 |

Table 4.2.4-1. (con't) WHC-SD-NR-CSER-010 Table 1 - Safe Masses and Associated Dimensions.

| | Sphere | | Hemisphere | | Infinite Cylinder | | Infinite Slab | |
|---|---|---|---|---|---|---|---|---|
| | Mass(k) | Dia. (cm) | Mass (k) | Dia (cm) | Mass (k/m) | Dia (cm) | Mass(k/m2) | Thick (cm) |
| MKIV Fuel Assemblies(0.95 wt% uranium inner and outer elements) | | | | | | | | |
| Billets | ----------Unlimited---------- | | | | | | | |
| Assemblies | 2371 | 87.6 | 3890 | 130.3 | 1741 | 57.4 | 1572 | 24.9 |
| Outer elements | 1635 | 84.3 | 2687 | 125.0 | 1249 | 55.1 | 1177 | 24.1 |
| Inner elements | 1399 | 77.7 | 2323 | 116.1 | 1147 | 50.5 | 1167 | 22.6 |
| Scrap | 975 | 67.6 | 1634 | 108.7 | 844 | 49.3 | 879 | 23.1 |
| Solution | ----------Unlimited---------- | | | | | | | |
| MKIA Fuel Assemblies(0.95 wt% uranium inner element, 1.25 wt% uranium outer element) | | | | | | | | |
| Billets | 4472 | 91.2 | 7243 | 53.2 | 3176 | 59.9 | 2851 | 25.4 |
| Assemblies | 724 | 61.2 | 1232 | 36.4 | 723 | 40.4 | 913 | 17.3 |
| Outer elements | 423 | 56.1 | 734 | 33.5 | 458 | 36.8 | 620 | 15.5 |
| Inner elements | 1441 | 78.0 | 2397 | 45.7 | 1153 | 52.8 | 1157 | 22.1 |
| Scrap | 266 | 51.1 | 470 | 30.7 | 310 | 34.0 | 430 | 15.0 |
| Solution | 682 | 71.4 | 1152 | 42.2 | 1204 | 46.2 | 635 | 22.6 |

Note that MKIA billets, scrap, and solution results are based on 1.25 wt% enrichment

## 4.3 TOOLS USED IN THE ANALYSIS

Hazard Analysis
GENII The Hanford Environmental Radiation Dosimetry Software
Criticality Study
Study of Combustion of Uranium
Fire Loading Study
Safety Class Guidelines
Fire Safety (Source of Standard Time-Temp Curve)

## 5.0 REFERENCES

DOE-RL, 1991, *303-K Radioactive Mixed-Waste Storage Facility Closure Plan*, DOE/RL-90-04, Rev. 1, U.S. Department of Energy Richland Field Office, Richland, Washington.

NRC, 1988, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, NUREG-1320, May 1988, U. S. Nuclear Regulatory Commission, Washington D. C.

WHC-CM-1-3, *Management Requirements and Procedures*, MRP 5.46, "Safety Classification of Systems, Components, and Structures," Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, 7.0, "Risk," Westinghouse Hanford Company, Richland, Washington.

WHC-CM-6-32, *Safety Analysis and Engineering Work Procedures*, WP-5.5, "Final Safety Analysis Reports," Westinghouse Hanford Company, Richland, Washington.

Besser, R. L., 1994, *Technical Safety Requirements for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-TSR-001, Westinghouse Hanford Company, Richland, Washington.

Brehm, J. R., and T. L. Deobald, 1994, *Interim Safety Basis for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-ISB-001, Westinghouse Hanford Company, Richland, Washington.

Gimera, R. J., 1992, *N Reactor Fuel Fabrication Facilities Shutdown Program Plan*, WHC-SP-0793, Westinghouse Hanford Company, Richland, Washington.

Hilliard, R. K., 1958, *Oxidation of Uranium in Air at High Temperatures*, HW-58022, General Electric, Hanford Atomic Products Operation, Richland, Washington.

Hooker, C. D., and D. F. Hadlock, J. Mishima, R. L. Gilchrist, 1983, *Hazard Classification Test of The Cartridge, 120 mm, APFSD-T, XM829*, PNL-4459, Battelle, Pacific Northwest Laboratory, Richland, Washington.

Huang, C. H., 1993, *Hazard Classification for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-HC-004, Rev 0, Westinghouse Hanford Company, Richland, Washington.

Johnson, D. J., and J. R. Brehm, 1994, *Hazard Analysis for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-PHA-002, Rev 0, Westinghouse Hanford Company, Richland, Washington.

Kelly, J. E., 1994, *Fire Criticality Probability Analysis for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-TI-051, Rev 0, Westinghouse Hanford Company, Richland, Washington.

Mishima, J., and M. A. Parkhurst, R. I. Scherpletz, D. E. Hadlock, 1985, *Potential Behavior of Depleted Uranium Penetrators under Shipping and Bulk Storage Accident Conditions*, PNL-5415, Battelle, Pacific Northwest Laboratory, Richland, Washington.

Myott, C. F., 1993, *Fire Loading Calculations for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-TI-053, Rev 0, Westinghouse Hanford Company, Richland, Washington.

Myott, C. F., 1994, *Fire Hazards Analysis for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-FHA-001, Rev 0, Westinghouse Hanford Company, Richland, Washington.

Schwinkendorf, K. N., 1993, *Criticality Safety Evaluation Report for 300 Area Fissile Material Limits and Fuel Storage*, WHC-SD-NR-CSER-010, Rev 0, Westinghouse Hanford Company, Richland, Washington.

This page intentionally left blank.

# APPENDIX A

# GENII ANALYSIS FOR FACILITY

This page intentionally left blank.

From: Radiological & Toxicological Analysis 29250-DAH-93010
Phone: 6-8190 H4-64
Date: April 28, 1993
Subject: RADIOLOGICAL CONSEQUENCES OF A RELEASE OF ONE METRIC TON OF URANIUM FROM THE N-FUELS PROCESSING AREA

To: D. J. Johnson H4-68

cc: R. G. Britton H4-64
J. S. Davis H4-64
J. C. Van Keuren H4-64
DAH File/LB

The subject analysis which you requested is attached.

If you have questions or further needs, feel free to call me at 6-8191.

D. A. Himes
Principal Engineer

raw

Concurrence: John C Van Keuren 4/28/93
John C. Van Keuren, Manager Date
Radiological & Toxicological Analysis

Attachment

# RADIOLOGICAL CONSEQUENCES OF A RELEASE OF ONE METRIC TON OF URANIUM FROM THE N-FUELS PROCESSING AREA

D.A. Himes
4/2/93

The radiological consequences of a unit release of 1 MT (approximately 1 Ci) of uranium of specified composition along with associated Tc-99 from the N-Fuels Processing Area are required [1]. The N-Fuels Processing Area is located in Buildings 333 and 3712 in the 300 Area. A previous analysis [2] determined radiological consequences for essentially the same release of uranium, but without the Tc-99 contribution.

Source Term Development:

The isotopic composition specified [1] for the 1.25% enriched uranium metal in the facility is shown in Table 1.

Table 1: Specified composition of 1 MT of uranium in the N-Fuels Processing Area [1]

| Isotope | % of U metal | grams per MT U | Ci per MT U |
|---|---|---|---|
| U 234 | 0.009 | 9.27E+1 | 5.8E-1 |
| U 235 | 1.25 | 1.30E+4 | 2.7E-2 |
| U 236 | 0.069 | 7.2E+2 | 4.7E-2 |
| U 238 | 98.7 | 1.0E+6 | 3.5E-1 |
| Tc 99 | | 1.0E+1 | 1.7E-1 |

Transport Assumptions:

The N-Fuels Processing Area is located near the eastern edge of the Hanford Reservation where the nearest site boundary is the adjacent river. Where the site is bounded by the Columbia River, the site boundary is taken to be at the nearer bank of the river for purposes of estimating inhalation and submersion doses. The corresponding agricultural area (residence of the Ingestion Pathway Receptor) is assumed to be on the east bank of the river.

The 95 percentile worst-case dispersion factor ($\chi/Q$) was calculated by the GENII code based on 300 Area meteorology data over the period 1983 to 1987. Depending on scenario, the source could have a considerable size (up to about 80 m across). Due to the uncertainty in source size, no credit was taken for the effect of source size or plume meander.

Receptor Descriptions:

Onsite:

Normally, for ground level releases, the receptor at a distance of 100 m in the worst direction [3]. Doses calculated for this receptor include inhalation and submersion. The maximum onsite receptor in this case is 100 m east with an acute 95 percentile sector X/Q = 3.4E-2 $s/m^3$ [4].

Site Boundary:

Receptor at the site boundary in the worst direction. This receptor is assumed to stay at this location for the duration of the accident. Doses calculated include inhalation and submersion. In this case, the maximum site boundary receptor is 490 m east with an acute 95 percentile sector X/Q = 2.3E-3 $s/m^3$.

Agricultural Area:

Residence of the ingestion pathway receptor (IPR). This receptor is assumed to grow his own food, including a variety of crops, meat and dairy products and to continue to do so at this location for 50 years following the accident. No credit is taken for uncontaminated foodstuffs brought in from outside the area. Note that IPR ingestion doses are reported only as a measure of economic damage since, in the case of an accident, any contaminated land or products would not be used. Ingestion and ground shine would not, therefore, be actual exposure pathways. In this case, the maximum IPR is 1140 m east with an acute 95 percentile sector X/Q = 5.5E-4 $s/m^3$. The release is normally assumed to occur just prior to the autumn harvest in order to maximize consequences with regard to the time of the accident. Since this analysis is to be used for a hazard classification, results for the winter scenario are included also.

Code Documentation:

GENII version 1.485 (12/3/90) [5]
RMDLIB - Radionuclide Master Library (11/15/90)
External Dose Factor Library (5/8/90)
Internal Dose Increment Library, Worst Case solubilities, (12/3/90 PDR)
Joint Frequency Data: 300 Area, 10 m, Pasquill A-F (1983-1987 Average)

Typical GENII input files are attached for reference.

Results:

Resulting doses for a 1 MT release of the mix shown in Table 1 are shown in Table 2.

Table 2: Resulting doses for a 1 MT release of the mix shown in Table 1

| Receptor | Dose Type | Dose (rem) EDE | Dose (rem) Limiting Organ |
|---|---|---|---|
| Onsite (100 m E) | Inhalation | 1.5E+3 | 1.2E+4 (Lung) |
| | Submersion | 1.9E-5 | 1.9E-5 |
| | * Total | 1.5E+3 | 1.2E+4 (Lung) |
| Site Boundary (490 m E) | Inhalation | 9.6E+1 | 8.0E+2 (Lung) |
| | Submersion | 1.3E-6 | 1.3E-6 |
| | * Total | 9.6E+1 | 8.0E+2 (Lung) |
| IPR (1140 m E) Autumn | Inhalation | 2.3E+1 | 1.9E+2 (Lung) |
| | | | 1.2E-2 (Bone Surf) |
| | Submersion | 3.1E-7 | 3.1E-7 |
| | Ingestion | 5.0E+0 | 1.0E-1 (Lung) |
| | | | 7.2E+1 (Bone Surf) |
| | Ground Shine | 3.3E-3 | 3.3E-3 |
| | * Total | 2.8E+1 | 1.9E+2 (Lung) |
| IPR (1140 m E) Winter | Inhalation | 2.3E+1 | 1.9E+2 (Lung) |
| | | | 1.2E-2 (Bone Surf) |
| | Submersion | 3.1E-7 | 3.1E-7 |
| | Ingestion | 2.7E-2 | 1.4E-3 (Lung) |
| | | | 2.3E-1 (Bone Surf) |
| | Ground Shine | 3.3E-3 | 3.3E-3 |
| | * Total | 2.3E+1 | 1.9E+2 (Lung) |

Note that only the winter ingestion EDE is compared to the environmental impact criteria in Reference 3. The EPA PPAG is 0.5 rem EDE and the EPA EPAG is 5 rem EDE [6].

References:

1. DSI, D.J. Johnson to D.A. Himes, no title, March 24, 1993 (attached).

2. Memo, D.A. Himes to D.J. Johnson, "Radiological Consequences of a Release of One Curie of Uranium From the N-Fuels Processing Area," 29250-DAH-92018, September 16, 1992.

3. Nonreactor Facility Safety Analysis Manual, Westinghouse Hanford Co., WHC-CM-4-46, September 18, 1992.

4. Memo, L.M. Deere to J.P. Hinckley and E.E. Leitz, "Documentation of Acute, Ground Level, 100 m Maximum Individual, X/Q Values," Radiological Safety Analysis, January 31, 1991.

5. B.A. Napier, et al., GENII - The Hanford Environmental Radiation Dosimetry Software System, PNL-6484, Dec. 1988.

6. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA 520/1-75-001A, January 1990.

Copy of Requesting Memo (Reference 1)

DON'T SAY IT --- Write it! March 24, 1993

TO: D. A. Himes
Phone: 6-8190

FROM: D. J. Johnson N1-37
Phone 6-9098

Dave,

It looks a though the easiest thing to do is simply add the 10 g of $^{99}Tc$ to the present GENII run. I made up this table, it deviates slightly from yours but the curies are the same.

| Composition of 1 MT Uranium Billet or Fuel Assembly | | | | |
|---|---|---|---|---|
| Isotope | %1 | g | Ci | % Ci |
| $^{234}U$ | 0.009% | 9.27E+1 | 0.58 | 58% |
| $^{235}U$ | 1.25% | 1.30E+4 | 0.027 | 2.7% |
| $^{236}U$ | 0.069% | 7.2E+2 | 0.047 | 4.7% |
| $^{238}U$ | 98.7% | 1.0E+6 | 0.346 | 34 |
| $^{99}Tc$ | | 10 | 0.17 | |
| Total | | 1 E6 | 1.17 | |

A copy of your memo is attached.

"TO MAKE LIFE LAST PUT SAFETY FIRST"

**GENII Input File**
**Onsite Receptor 100 m E**

```
######################## Program GENII Input File ############  8 Jul 88 ####
Title: N-Reactor fuel in 300 Area - OS 100 m E
       \GENII\nrfullos.in                          Created on 08-07-1992 at 13:41
OPTIONS========================= Default ===================================
F    Near-field scenario?      (Far-field)    NEAR-FIELD: narrowly-focused
F    Population dose?        (Individual)                 release, single site
T    Acute release?            (Chronic)    FAR-FIELD:  wide-scale release,
     Maximum Individual data set used                     multiple sites

                                  Complete                                Complete
TRANSPORT OPTIONS============== Section   EXPOSURE PATHWAY OPTIONS====== Section
T  Air Transport                    1      F  Finite plume, external        5
F  Surface Water Transport          2      T  Infinite plume, external      5
F  Biotic Transport (near-field)  3,4      F  Ground, external              5

F  Waste Form Degradation (near)  3,4      F  Recreation, external          5

                                           T  Inhalation uptake             5,6
REPORT OPTIONS========================     F  Drinking water ingestion      7,8
T  Report AEDE only                        F  Aquatic foods ingestion       7,8
T  Report by radionuclide                  F  Terrestrial foods ingestion   7,9
T  Report by exposure pathway              F  Animal product ingestion      7,10
F  Debug report on screen                  F  Inadvertent soil ingestion

INVENTORY ###########################################################################

 4   Inventory input activity units: (1-pCi  2-uCi  3-mCi  4-Ci  5-Bq)
 0   Surface soil source units (1- m2  2- m3  3- kg)
     Equilibrium question goes here

     --------|----Release Terms------|----------Basic Concentrations---------|
     Use when|  transport selected   |  near-field scenario, optionally       |
     --------|-----------------------|----------------------------------------|
     Release |        Surface Buried |        Surface Deep    Ground  Surface |

     Radio-  |Air     Water   Waste  |Air     Soil    Soil    Water   Water  |
     nuclide |/yr     /yr     /m3    |/m3     /unit   /m3     /L      /L     |
     --------|------ ------- -------|------- ------- ------- ------- -------|
     TC99    1.7E-01
     U 234   5.8E-01
     U 235   2.7E-02
     U 236   4.7E-02
     U 238   3.5E-01

     --------|----Derived Concentrations-----|
     Use when|   measured values are known   |
     --------|-------------------------------|
```

| Release Radio-nuclide | Terres. Plant /kg | Animal Product /kg | Drink Water /L | Aquatic Food /kg |
|---|---|---|---|---|

TIME ##########################################################################

1 Intake ends after (yr)
50 Dose calc. ends after (yr)
0 Release ends after (yr)
0 No. of years of air deposition prior to the intake period
0 No. of years of irrigation water deposition prior to the intake period

FAR-FIELD SCENARIOS (IF POPULATION DOSE) ####################################

0 Definition option: 1-Use population grid in file POP.IN
0 2-Use total entered on this line

NEAR-FIELD SCENARIOS ########################################################

Prior to the beginning of the intake period: (yr)
0 When was the inventory disposed? (Package degradation starts)
0 When was LOIC? (Biotic transport starts)
0 Fraction of roots in upper soil (top 15 cm)
0 Fraction of roots in deep soil
0 Manual redistribution: deep soil/surface soil dilution factor
0 Source area for external dose modification factor (m2)

TRANSPORT ###################################################################

====AIR TRANSPORT==========SECTION 1=====

Option: 0-Calculate PM, 1-Use chi/Q or PM value, 2-Select MI dist & dir, 3-Specify MI dist & dir

3 Option: 1-Use chi/Q or PM value
0 Chi/Q or PM value
13 MI sector index (1=S)
100.0 MI distance from release point (m)
T Use jf data, (T/F) else chi/Q grid

0 Release type (0-3)
F Stack release (T/F)
0 Stack height (m)
0 Stack flow (m3/sec)
0 Stack radius (m)
0 Effluent temp. (C)
0 Building x-section (m2)
0 Building height (m)

====SURFACE WATER TRANSPORT==========SECTION 2=====

0 Mixing ratio model: 0-use value, 1-river, 2-lake
0 Mixing ratio, dimensionless
0 Average river flow rate for: MIXFLG=0 (m3/s), MIXFLG=1,2 (m/s),
0 Transit time to irrigation withdrawl location (hr)
If mixing ratio model > 0:
0 Rate of effluent discharge to receiving water body (m3/s)
0 Longshore distance from release point to usage location (m)
0 Offshore distance to the water intake (m)
0 Average water depth in surface water body (m)
0 Average river width (m), MIXFLG=1 only
0 Depth of effluent discharge point to surface water (m), lake only

====WASTE FORM AVAILABILITY==========SECTION 3=====

```
0          Waste form/package half life, (yr)
0          Waste thickness, (m)
0          Depth of soil overburden, m

           ----BIOTIC TRANSPORT OF BURIED SOURCE----------SECTION 4-----
T          Consider during inventory decay/buildup period (T/F)?
T          Consider during intake period (T/F)?    | 1-Arid non agricultural
0          Pre-Intake site condition.....          | 2-Humid non agricultural
                                                   | 3-Agricultural

EXPOSURE ##################################################################

           ----EXTERNAL EXPOSURE--------------------------------SECTION 5-----
           Exposure time:                  |  Residential irrigation:
0            Plume (hr)                    | T       Consider: (T/F)
0            Soil contamination (hr)       | 0       Source: 1-ground water
0            Swimming (hr)                 |                 2-surface water
0            Boating (hr)                  | 0       Application rate (in/yr)
0            Shoreline activities (hr)     | 0       Duration (mo/yr)
0          Shoreline type: (1-river, 2-lake, 3-ocean, 4-tidal basin)
0          Transit time for release to reach aquatic recreation (hr)
1.0        Average fraction of time submersed in acute cloud (hr/person hr)

           ----INHALATION---------------------------------------SECTION 6-----
8766.0     Hours of exposure to contamination per year
0          0-No resus-  1-Use Mass Loading          2-Use Anspaugh model
0            pension      Mass loading factor (g/m3)   Top soil available (cm)

           ----INGESTION POPULATION-----------------------------SECTION 7-----
0          Atmospheric production definition (select option):
0            0-Use food-weighted chi/Q, (food-sec/m3), enter value on this
line
             1-Use population-weighted chi/Q
             2-Use uniform production
             3-Use chi/Q and production grids (PRODUCTION will be overridden)
0          Population ingesting aquatic foods, 0 defaults to total (person)
0          Population ingesting drinking water, 0 defaults to total (person)
F          Consider dose from food exported out of region (default=F)

           Note below: S* or Source: 0-none, 1-ground water, 2-surface water
                                     3-Derived concentration entered above
           ---- AQUATIC FOODS / DRINKING WATER INGESTION---------SECTION 8----

F          Salt water? (default is fresh)

           USE        TRAN-  PROD-    -CONSUMPTION-  |
           ?   FOOD   SIT    UCTION   HOLDUP   RATE  |
           T/F TYPE   hr     kg/yr    da       kg/yr |      DRINKING WATER
           --- ------ -----  -------  ------   -----  |-------------------------
           F   FISH    0.00  0.0E+00   0.00     0.0  | 0      Source (see
above)
```

F MOLLUS 0.00 0.0E+00 0.00 0.0 | T Treatment? T/F

F CRUSTA 0.00 0.0E+00 0.00 0.0 | 0
Holdup/transit(da)

F PLANTS 0.00 0.0E+00 0.00 0.0 | 0 Consumption
(L/yr)

====TERRESTRIAL FOOD INGESTION==========SECTION 9=====

| USE ? T/F | FOOD TYPE | GROW TIME da | --IRRIGATION-- S * | RATE in/yr | TIME mo/yr | YIELD kg/m2 | PROD-UCTION kg/yr | --CONSUMPTION-- HOLDUP da | RATE kg/yr |
|---|---|---|---|---|---|---|---|---|---|
| F | LEAF V | 0.00 | 0 | 0.0 | 0.0 | 0.0 | 0.0E+00 | 0.0 | 0.0 |
| F | ROOT V | 0.00 | 0 | 0.0 | 0.0 | 0.0 | 0.0E+00 | 0.0 | 0.0 |
| F | FRUIT | 0.00 | 0 | 0.0 | 0.0 | 0.0 | 0.0E+00 | 0.0 | 0.0 |
| F | GRAIN | 0.00 | 0 | 0.0 | 0.0 | 0.0 | 0.0E+00 | 0.0 | 0.0 |

====ANIMAL PRODUCTION CONSUMPTION==========SECTION 10====

| USE ? T/F | FOOD TYPE | ---HUMAN---- CONSUMPTION RATE kg/yr | HOLDUP da | TOTAL PROD-UCTION kg/yr | DRINK WATER CONTAM FRACT. | STORED FEED DIET FRAC-TION | GROW TIME da | -IRRIGATION-- S * | RATE in/yr | TIME mo/yr | YIELD kg/m3 | STOR-AGE da |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| F | BEEF | 0.0 | 0.0 | 0.00 | 0.00 | 0.00 | 0.0 | 0 | 0.0 | 0.00 | 0.00 | 0.0 |
| F | POULTR | 0.0 | 0.0 | 0.00 | 0.00 | 0.00 | 0.0 | 0 | 0.0 | 0.00 | 0.00 | 0.0 |
| F | MILK | 0.0 | 0.0 | 0.00 | 0.00 | 0.00 | 0.0 | 0 | 0.0 | 0.00 | 0.00 | 0.0 |
| F | EGG | 0.0 | 0.0 | 0.00 | 0.00 | 0.00 | 0.0 | 0 | 0.0 | 0.00 | 0.00 | 0.0 |
| | | | | | | FRESH FORAGE | | | | | | |
| | BEEF | | | | | 0.00 | 0.0 | 0 | 0.0 | 0.00 | 0.00 | 0.0 |
| | MILK | | | | | 0.00 | 0.0 | 0 | 0.0 | 0.00 | 0.00 | 0.0 |

##########

**GENII Input File**
**Ingestion Pathway Receptor 1140 m E**

```
######################## Program GENII Input File ########### 8 Jul 88 ####
Title: N-Reactor fuel in 300 Area - IPR 1140 m E
       \GENII\nrfullip.in                        Created on 08-07-1992 at 13:45
OPTIONS---------------------- Default ------------------------------------
F    Near-field scenario?      (Far-field)    NEAR-FIELD: narrowly-focused
F    Population dose?         (Individual)                release, single site
T    Acute release?             (Chronic)    FAR-FIELD:  wide-scale release,
     Maximum Individual data set used                     multiple sites

                                  Complete                                Complete
TRANSPORT OPTIONS------------     Section   EXPOSURE PATHWAY OPTIONS-----  Section
T  Air Transport                     1      F  Finite plume, external        5
F  Surface Water Transport           2      F  Infinite plume, external      5
F  Biotic Transport (near-field)    3,4     T  Ground, external              5

F  Waste Form Degradation (near)    3,4     F  Recreation, external          5

                                            F  Inhalation uptake             5,6
REPORT OPTIONS----------------------        F  Drinking water ingestion      7,8
F  Report AEDE only                         F  Aquatic foods ingestion       7,8
T  Report by radionuclide                   T  Terrestrial foods ingestion   7,9
T  Report by exposure pathway               T  Animal product ingestion      7,10
F  Debug report on screen                   T  Inadvertent soil ingestion

INVENTORY ##########################################################################

 4   Inventory input activity units: (1-pCi  2-uCi  3-mCi  4-Ci  5-Bq)
 0   Surface soil source units (1- m2  2- m3  3- kg)
     Equilibrium question goes here

     --------|----Release Terms------|----------Basic Concentrations---------|
     Use when|  transport selected   |  near-field scenario, optionally       |
     --------|-----------------------|----------------------------------------|
     Release |        Surface Buried |        Surface Deep    Ground  Surface |

     Radio-  |Air     Water   Waste  |Air     Soil    Soil    Water   Water   |
     nuclide |/yr     /yr     /m3    |/m3     /unit   /m3     /L      /L      |
     --------|------- ------- -------|------- ------- ------- ------- -------|
     TC99    1.7E-01
     U 234   5.8E-01
     U 235   2.7E-02
     U 236   4.7E-02
     U 238   3.5E-01

     --------|----Derived Concentrations-----|
     Use when|   measured values are known   |
     --------|-------------------------------|
```

| Release Radio-nuclide | Terres. Plant /kg | Animal Product /kg | Drink Water /L | Aquatic Food /kg |
|---|---|---|---|---|

TIME ##########################################################################

50 Intake ends after (yr)
50 Dose calc. ends after (yr)
0 Release ends after (yr)
0 No. of years of air deposition prior to the intake period
0 No. of years of irrigation water deposition prior to the intake period

FAR-FIELD SCENARIOS (IF POPULATION DOSE) ##################################

0 Definition option: 1-Use population grid in file POP.IN
0 2-Use total entered on this line

NEAR-FIELD SCENARIOS ######################################################

Prior to the beginning of the intake period: (yr)
0 When was the inventory disposed? (Package degradation starts)
0 When was LOIC? (Biotic transport starts)
0 Fraction of roots in upper soil (top 15 cm)
0 Fraction of roots in deep soil
0 Manual redistribution: deep soil/surface soil dilution factor
0 Source area for external dose modification factor (m2)

TRANSPORT ##################################################################

----AIR TRANSPORT----------SECTION 1-----

| | | | |
|---|---|---|---|
| | 0-Calculate PM | 0 | Release type (0-3) |
| 3 | Option: 1-Use chi/Q or PM value | F | Stack release (T/F) |
| | 2-Select MI dist & dir | 0 | Stack height (m) |
| | 3-Specify MI dist & dir | 0 | Stack flow (m3/sec) |
| 0 | Chi/Q or PM value | 0 | Stack radius (m) |
| 13 | MI sector index (1=S) | 0 | Effluent temp. (C) |
| 1140.0 | MI distance from release point (m) | 0 | Building x-section (m2) |
| T | Use jf data, (T/F) else chi/Q grid | 0 | Building height (m) |

----SURFACE WATER TRANSPORT----------SECTION 2-----

0 Mixing ratio model: 0-use value, 1-river, 2-lake
0 Mixing ratio, dimensionless
0 Average river flow rate for: MIXFLG=0 (m3/s), MIXFLG=1,2 (m/s),
0 Transit time to irrigation withdrawl location (hr)
If mixing ratio model > 0:
0 Rate of effluent discharge to receiving water body (m3/s)
0 Longshore distance from release point to usage location (m)
0 Offshore distance to the water intake (m)
0 Average water depth in surface water body (m)
0 Average river width (m), MIXFLG=1 only
0 Depth of effluent discharge point to surface water (m), lake only

----WASTE FORM AVAILABILITY----------SECTION 3-----

```
0          Waste form/package half life, (yr)
0          Waste thickness, (m)
0          Depth of soil overburden, m

           ----BIOTIC TRANSPORT OF BURIED SOURCE----------SECTION 4-----
T          Consider during inventory decay/buildup period (T/F)?
T          Consider during intake period (T/F)?    | 1-Arid non agricultural
0          Pre-Intake site condition.....          | 2-Humid non agricultural
                                                   | 3-Agricultural

EXPOSURE ##################################################################

           ----EXTERNAL EXPOSURE--------------------------------SECTION 5-----
           Exposure time:                  |  Residential irrigation:
0            Plume (hr)                    | T       Consider: (T/F)
4380.0       Soil contamination (hr)       | U       Source: 1-ground water
0            Swimming (hr)                 |                 2-surface water
0            Boating (hr)                  | 0       Application rate (in/yr)
0            Shoreline activities (hr)     | 0       Duration (mo/yr)
0          Shoreline type: (1-river, 2-lake, 3-ocean, 4-tidal basin)
0          Transit time for release to reach aquatic recreation (hr)
1.0        Average fraction of time submersed in acute cloud (hr/person hr)

           ----INHALATION---------------------------------------SECTION 6-----
8766.0     Hours of exposure to contamination per year
0          0-No resus-  1-Use Mass Loading          2-Use Anspaugh model
0            pension      Mass loading factor (g/m3)   Top soil available (cm)

           ----INGESTION POPULATION-----------------------------SECTION 7-----
 1         Atmospheric production definition (select option):
0            0-Use food-weighted chi/Q, (food-sec/m3), enter value on this
line
             1-Use population-weighted chi/Q
             2-Use uniform production
             3-Use chi/Q and production grids (PRODUCTION will be overridden)
0          Population ingesting aquatic foods, 0 defaults to total (person)
0          Population ingesting drinking water, 0 defaults to total (person)
F          Consider dose from food exported out of region (default=F)

           Note below: S* or Source: 0-none, 1-ground water, 2-surface water
                                     3-Derived concentration entered above
           ---- AQUATIC FOODS / DRINKING WATER INGESTION---------SECTION 8----

F          Salt water? (default is fresh)

           USE        TRAN-  PROD-    -CONSUMPTION-  |
           ?   FOOD   SIT    UCTION   HOLDUP   RATE  |
           T/F TYPE   hr     kg/yr    da       kg/yr |      DRINKING WATER
           --- ------ -----  -------  ------   -----  |------------------------
           F   FISH    0.00  0.0E+00   0.00     0.0  | 0      Source (see
above)
```

```
          F   MOLLUS  0.00  0.0E+00    0.00     0.0 | T      Treatment? T/F

          F   CRUSTA  0.00  0.0E+00    0.00     0.0 | 0
Holdup/transit(da)
          F   PLANTS  0.00  0.0E+00    0.00     0.0 | 0      Consumption
(L/yr)
```

----TERRESTRIAL FOOD INGESTION----------SECTION 9-----

| USE ? T/F | FOOD TYPE | GROW TIME da | S * | --IRRIGATION-- RATE in/yr | TIME mo/yr | YIELD kg/m2 | PROD-UCTION kg/yr | --CONSUMPTION-- HOLDUP da | RATE kg/yr |
|---|---|---|---|---|---|---|---|---|---|
| T | LEAF V | 90.00 | 0 | 0.0 | 0.0 | 1.5 | 0.0E+00 | 1.0 | 30.0 |
| T | ROOT V | 90.00 | 0 | 0.0 | 0.0 | 4.0 | 0.0E+00 | 5.0 | 220.0 |
| T | FRUIT | 90.00 | 0 | 0.0 | 0.0 | 2.0 | 0.0E+00 | 5.0 | 330.0 |
| T | GRAIN | 90.00 | 0 | 0.0 | 0.0 | 0.8 | 0.0E+00 | 180.0 | 80.0 |

----ANIMAL PRODUCTION CONSUMPTION----------SECTION 10----

| | | ---HUMAN---- | | TOTAL | DRINK | ----------STORED FEED---------- | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| USE ? T/F | FOOD TYPE | CONSUMPTION RATE kg/yr | HOLDUP da | PROD-UCTION kg/yr | WATER CONTAM FRACT. | DIET FRAC-TION | GROW TIME da | -IRRIGATION-- S * | RATE in/yr | TIME mo/yr | YIELD kg/m3 | STOR-AGE da |
| T | BEEF | 80.0 | 15.0 | 0.00 | 0.00 | 0.25 | 90.0 | 0 | 0.0 | 0.00 | 0.80 | 0.0 |
| T | POULTR | 18.0 | 1.0 | 0.00 | 0.00 | 1.00 | 90.0 | 0 | 0.0 | 0.00 | 0.80 | 0.0 |
| T | MILK | 270.0 | 1.0 | 0.00 | 0.00 | 0.25 | 45.0 | 0 | 0.0 | 0.00 | 2.00 | 0.0 |
| T | EGG | 30.0 | 1.0 | 0.00 | 0.00 | 1.00 | 90.0 | 0 | 0.0 | 0.00 | 0.80 | 0.0 |
| | | | | | | ----------FRESH FORAGE---------- | | | | | | |
| | BEEF | | | | | 0.75 | 45.0 | 0 | 0.0 | 0.00 | 2.00 | 100.0 |
| | MILK | | | | | 0.75 | 30.0 | 0 | 0.0 | 0.00 | 1.50 | 0.0 |

##########

## CHECKLIST FOR PEER REVIEW

Document Reviewed: **RADIOLOGICAL CONSEQUENCES OF A RELEASE OF ONE METRIC TON OF URANIUM FROM THE N-FUELS PROCESSING AREA, D.A. Himes, 4/2/93**

Scope of Review: entire document

| Yes | No | NA | | |
|---|---|---|---|---|
| [X] | [ ] | [ ] | * | Previous reviews complete and cover analysis, up to scope of this review, with no gaps. |
| [X] | [ ] | [ ] | | Problem completely defined. |
| [ ] | [ ] | [X] | 1. | Accident scenarios developed in a clear and logical manner. |
| [X] | [ ] | [ ] | | Necessary assumptions explicitly stated and supported. |
| [X] | [ ] | [ ] | | Computer codes and data files documented. |
| [X] | [ ] | [ ] | | Data used in calculations explicitly stated in document. |
| [X] | [ ] | [ ] | | Data checked for consistency with original source information as applicable. |
| [X] | [ ] | [ ] | | Mathematical derivations checked including dimensional consistency of results. |
| [X] | [ ] | [ ] | | Models appropriate and used within range of validity or use outside range of established validity justified. |
| [X] | [ ] | [ ] | | Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations. |
| [X] | [ ] | [ ] | | Software input correct and consistent with document reviewed. |
| [X] | [ ] | [ ] | | Software output consistent with input and with results reported in document reviewed. |
| [X] | [ ] | [ ] | | Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references. |
| [ ] | [ ] | [X] | | Safety margins consistent with good engineering practices. |
| [X] | [ ] | [ ] | | Conclusions consistent with analytical results and applicable limits. |
| [X] | [ ] | [ ] | | Results and conclusions address all points required in the problem statement. |
| [ ] | [ ] | [X] | | Format consistent with appropriate NRC Regulatory Guide or other standards |
| [X] | | [ ] | * | Review calculations, comments, and/or notes are attached. |
| [X] | [ ] | [ ] | | Document approved. |

R. G. Britton R. G. Britton 4/13/93

Reviewer (Printed Name and Signature) Date

* Any calculations, comments, or notes generated as part of this review should be signed, dated and attached to this checklist. Such material should be labeled and recorded in such a manner as to be intelligible to a technically qualified third party.

[X] [ ] [ ] Analysis entered ~~into~~ analysis database

D.A. Himes 4/27/93

(Printed Name and Signature) Date

1. Customer specified Source term

**HEDOP REVIEW CHECKLIST**
**for**
**Radiological and Nonradiological Release Calculations**

Document reviewed (include title or description of calculation, document number, author, and date, as applicable):

**RADIOLOGICAL CONSEQUENCES OF A RELEASE OF ONE METRIC TON OF URANIUM FROM THE N-FUELS PROCESSING AREA**, D.A. Himes, 4/2/93

Submitted by: D.A. Himes Date Submitted: 4/22/93

Scope of Review: entire document

| YES | NO* | N/A | | |
|---|---|---|---|---|
| [X] | [ ] | [ ] | 1. | A detailed technical review and approval of the environmental transport and dose calculation portion of the analysis has been performed and documented. |
| [ ] | [X] | [ ] | 2. | Detailed technical review(s) and approval(s) of scenario and release determinations have been performed and documented. |
| [X] | [ ] | [ ] | 3. | HEDOP-approved code(s) were used. |
| [ ] | [ ] | [X] | 4. | Receptor locations were selected according to HEDOP recommendations. |
| [X] | [ ] | [ ] | 5. | All applicable environmental pathways and code options were included and are appropriate for the calculations. |
| [X] | [ ] | [ ] | 6. | Hanford site data were used. |
| [ ] | [ ] | [X] | 7. | Model adjustments external to the computer program were justified and performed correctly. |
| [X] | [ ] | [ ] | 8. | The analysis is consistent with HEDOP recommendations. |
| [ ] | | [X] | 9. | Supporting notes, calculations, comments, comment resolutions, or other information is attached. (Use the "Page 1 of X" page numbering format and sign and date each added page.) |
| [X] | [ ] | | 10. | **Approval is granted on behalf of the Hanford Environmental Dose Overview Panel.** |

* All "NO" responses must be explained and use of nonstandard methods justified.

Janet S. Davis Janet S. Davis 4/22/93
HEDOP-Approved Reviewer (Printed Name and Signature) Date

**COMMENTS (add additional signed and dated pages if necessary):**

2. The source term was specified by the customer. No evidence of review or approval was provided.
4. There are currently no HEDOP recommendations regarding receptor location.

# APPENDIX B

# OXIDATION OF URANIUM

This page intentionally left blank.

# OXIDATION OF URANIUM

## 1.0 INTRODUCTION

A literature search was conducted on uranium metal and Zircaloy-2 to determine their resulting oxidation characteristics when exposed to high temperatures associated with fire. Several experiments were identified as having results that can be applied to the material stored in the 3712 Building of the 300 Area N Reactor Fuel Fabrication and Storage Facility (Facility). The experiments were used to determine: uranium oxidation temperature, uranium billet oxidation time, uranium ignition temperature, and the effects of Zircaloy-2 cladding on ignition temperature.

## 2.0 OXIDATION STUDIES

### 2.1 DEPLETED URANIUM PENETRATOR TESTS

A test (Mishima, et al., 1985) was conducted for a safety analysis report on the depleted uranium penetrators (DU pens) portion of the 120 mm (4.72 in.) anti-tank missiles designed to break the armor of a tank. The missile consists of a DU pens with a steel-tipped aluminum windshield (nose cone), a sabot (shoe) to align the missile with the throat of the cannon, tail fins, casing, primer, and propellant. The sabot comes apart after leaving the cannon. To verify the safety during shipment, the missiles were subjected to various fire conditions. The 120 mm (4.72 in.) missiles, enclosed in shipping boxes, were placed on an iron framework made of 7.6 cm (3 in.) angle iron, 79 cm H x 61 cm W x 107 cm L (2.6 ft H x 2 ft W x 3.5 ft L) sitting in a 9 m x 9 m (30 ft x 30 ft) rimmed 7.6 cm (3 in.) steel tray covered with earth to the rim (to aid recovery of fragments). A large number of approximately 15 cm x 20 cm x 244 cm (6 in. x 8 in. x 8 ft) pieces of lumber (green railroad ties) were piled under, on top of, and around the iron framework. Then, 208 l (55 gal) of diesel fuel was sprayed on the pile to evenly coat the wood. The pile was then ignited remotely by a squib igniting 3.8 l (1 gal) of gas. The 120 mm (4.72 in.) missiles had a casing which allowed the propellant to "cook off," instead of firing. The fire burned vigorously during the first 3 to 4 hrs, with temperatures reaching 800 °C to 1100 °C (1472 °F to 2012 °F), and localized areas to 1200 °C (2192 °F). The railroad tie fire collapsed after about 4 hours, and the temperature dropped to around 600 °C (1112 °F) for about 12 hrs. The embers continued to emit heat, and about 24 hours after the ignition of the pile, the temperature was still about 350 °C (662 °F), and the material covered about 3 m x 3 m (10 ft x 10 ft). The temperature was still around 300 °C (572 °F) after 48 hrs when material recovery was started. The penetrators were not believed to have ignited, however, oxidation had continued in some cases even after recovery. The oxidation was concluded to be between 83 and 88% complete after 48 hrs.

The fire loading for the DU pens was 6 E+4 Cal wood/g DU pen (1.1 E+5 BTU wood/lb DU pen). In comparison, the free combustibles in the 3712 Building, from the fire loading survey (Myott 1993) in Table B-2, are only 4.6 E+2 Cal wood/g uranium (830 BTU wood/lb uranium). From this comparison, the uranium billet and fuel assembly wooden storage boxes comprise only 0.77% of

the fuel supply per unit weight that the DU pens had in the free-burn tests, indicating much lower oxidation temperatures and a shorter cooldown time for the 3712 Building. Additional fuel would be needed to completely oxidize the uranium billets and fuel pieces in the 3712 Building. While a fire in the 3712 Building might reach a temperature of 1093 °C (2000) °F, the free combustibles in the 3712 Building will be consumed in approximately 4 hrs (Myott 1993), about the same as for the DU pens tests. The collapsed fire stayed hot in the DU pens tests, because of the high concentration of combustibles, and the latent heat and insulation effect of the ashes.

The DU pens tests used diesel and gasoline to get the fire started. The material in the 3712 Building having no volatiles, will be more difficult to ignite. Also, the propagation time to all parts of the building will be relatively larger.

## 2.1 SMALL-PIECE URANIUM TESTS

Early studies (Hilliard 1958), of oxidation of uranium were conducted to determine how long it would take to produce uranium dioxide, $UO_2$, for use as reactor fuel. These tests were performed with small pieces (chips) of uranium in a furnace. The equations for burn time, were derived empirically from the tests to determine the time it takes to oxidize a uranium sample (Table B-1, Part 1A). The tests were conducted in a furnace where the specimen was heated 2 in argon, then, after reaching the test temperature, air flow (500 cm/min) was initiated. The temperature remained constant throughout the test. These equations were derived at four temperatures; 805 °C, 995 °C, 1200 °C, and 1440 °C (1481 °F, 1823 °F, 2192 °F, and 2624 °F).

The empirical equations were used to calculate the time it would take to oxidize a DU pens and an average uranium billet (Table B-1, Part 1B). Figure B-1 depicts the test data and the calculated data.

Some error is incurred when comparing a variable temperature fire to a constant temperature furnace. Delivering an oxygen supply after the pieces have reached test temperature, does not allow for the buildup of heat in an open oxidation. Also, the steady supply of oxygen eliminated the fluctuation of external temperature an open oxidation would have. Constant external temperature did not eliminate the fluctuation in internal temperature, as shown in Figures B-2 and B-3 (which are traced from Hilliard 1958, Figures 11 and 12, because data was not available in the text). Between about 400 °C and 650 °C (752 °F and 1202 °F), temperature variations occurred due to the thermal cycling effect caused by heat of reaction.

Table B-1. Oxidation Time For Uranium Test Specimens With Calculated Time For a DU Penetrator and an Average Billet. (Hilliard 1958)

| 1A. TEST AND CALCULATED OXIDATION TIME FOR URANIUM TEST SPECIMEN | | | | | |
|---|---|---|---|---|---|
| | EMPIRICAL EQUATIONS BY FURNACE TEMPERATURE | | | | |
| | $t_c$[a] = | $0.071(W_o/A_o)-73$ at 805 °C | $0.071(W_o/A_o)-76$ at 995 °C | $0.030(W_o/A_o)-33$ at 1200 °C | $0.015(W_o/A_o)-17$ at 1440 °C |
| TEST SPECIMEN GROUPS | $W_o/A_o$[b] ($mg/cm^2$) $W_o$ (mg) | $t_c$[a] (min) | $t_c$[a] (min) | $t_c$[a] (min) | $t_c$[a] (min) |
| A 0.323 cm dia 2.54 cm ht | 1437 4.0 E+3 | 60.0 | 25.0 | 11.5 | 6 |
| B 0.635 cm dia 1.9 cm ht | 2593 1.15 E+4 | 140 | 110 | 45 | 21.5 |
| C 1.31 cm dia 1.9 cm ht | 4639 m 4.86 E+4 | 280 | 245 | 105 | 50 |
| D 0.635 x 0.635 x 1.9 cm | 2591 1.48 E+4 | 140 | 110 | 45 | 21.5 |
| 1B. CALCULATED OXIDATION TIME FOR A DU PENETRATOR AND AN AVERAGE BILLET | | | | | |
| 2.79 cm dia 34.5 cm ht DU Penetrator | 12714 4.0 E+6 | 860 (14.3 hr) | 827 (13.7 hr) | 356 (6.2 hr) | 176.2 (2.9 hr) |
| Average Billet (See Fig. B-7) | 43476 1.39 E+8 | 3043.8 (50.7 hr) | 3010.7 (50.2 hr) | 1297.3 (21.6 hr) | 643.8 (10.7 hr) |

a. $t_c$, Time required to completely oxidize the specimen.
b. $W_o/A_o$, Starting Weight mg/Surface Area $cm^2$; $W_o$, starting weight is listed below $W_o/A_o$.

Variation in the oxide layer with furnace temperature was also noted. Below 300 °C (572 °F), $UO_2$ was produced. Above 300 °C (572 °F), $U_3O_8$ was indicated by the weight ratio of the oxygen to uranium. From 450 °C to about 535 °C (842 °F to 995 °F), the oxide was a fine black non-adherent powder. At 535 °C (995 °F), for tests under 60 minutes, the oxide was the same, but in runs of longer duration the oxide was granular. At 700 °C (1292 °F) and above, only hard black scale was produced. This was consistent with temperatures reached during the thermal cycles. Oxide formed during the high thermal peaks was sintered. At temperatures above the melting point, the oxide was extremely dense requiring a chisel to cleave it, and the residual was found as a dense teardrop-shaped nugget within the dense shell. There was no discussion in the reference of the effects of these changes on possible releases.

To find a total oxidation time for the billets in the 3712 Building, a comparison of the DU pens and the small-piece uranium tests was made. Using the tests results for the recovery of the DU pens where temperature was approximately 300 °C (572 °F) after 48 hours, and comparing the oxidation time

calculated for a DU pen using the empirical equations at 805 °C (1481 °F), a ratio can be found. The calculated time for the DU pens was 14.3 hrs at 805 °C (1481 °F), (Table B-1, Part 1B), giving a ratio of 48 hr/14.3 hr = 3.4. Using this ratio and applying it to the average billet calculated oxidation time at the same temperature (Table B-1, Part 1B), gives 3.4 x 50.7 hr = 172 hr as a comparable total billet oxidation time at 805 °C (1481 °F). At 995 °C (1823 °F), 3.4 x 50.2 hr = 171 hr total billet oxidation time. At 1200 °C (2192 °F), 3.4 x 21.6 hr = 73 hr total billet oxidation time. At 1440 °C (2624 °F), 3.4 x 10.7 hr = 36 hr total billet oxidation time. Comparing these total billet oxidation times calculated at the various temperatures with the estimated maximum temperature of 1093.3 °C (2000 °F)[1] for the combustibles fire in the 3712 Building, the extrapolated total billet oxidation time becomes about 124 hours. To be conservative an estimate of 100 hrs for complete uranium billet oxidation will be used.

Note: The 3712 Building has a combustible temperature of 1093 °C (2000 °F), while the 303-B and 303-G Buildings have temperatures of 1166 °C (2131°F) and 1156 °C (2113 °F), respectively. Based on a fire temperature of 1093 °C (2000 °F) the extrapolated estimated average billet total oxidation time is 124 hours and a 100 hour total oxidation time, 81%, was used for conservatism. For a fire temperatures of 1166 °C (2131 °F) the extrapolated estimated average billet total oxidation time is 89 hours and applying the same conservatism the total oxidation time is 72 hours. For the 8 hour fire the release fraction from the 3712 Building is 8/100 X 1122 MTU or 89.9 MTU. For the 303-B Building the release fraction is 8/72 X 400 MTU (the defined capacity) or 44.4 MTU. Therefore, the 3712 Building fire is the bounding case.

---

[1]The enthalpy of formation associated with the uranium that is oxidized in the 3712 Building represents about 35 percent of the free combustibles in the building. This additional BTU loading has the potential for increasing the fire temperature. The empirical relationships that were used to estimate billet oxidation times likely also did not include the uranium oxidation heat of reaction and the quantity of uranium oxidized with respect to the free combustibles was lower. No effort was made to quantify the temperature effect but feel that this is encompassed by the other conservatisms associated with the analysis.

Table B-2. Comparisons Between the Free-Burn of the 120 mm Missile DU Penetrators and a Fire in the 3712 Building.

| 3712 BUILDING | | | | | |
|---|---|---|---|---|---|
| L | W | Area | Fire Loading from Boxes and Plastic. | Uranium Stored 1122 MTU | |
| ft | ft | $ft^2$ | Btu | # U | BTU W/# U |
| 90 | 108 | 9720 | 2.05 E+9 | 2.47 E+6 | 8.30 E+2 |
| m | m | $m^2$ | Cal | g U | Cal W/g U |
| 27.4 | 32.9 | 903.0 | 5.17 E+11 | 1.122 E+9 | 4.61 E+2 |
| PENETRATOR TEST | | | | | |
| L | W | Area | Fire Loading* from 2 Cords Wood and 11440 L (55 gal) Diesel Fuel | 12 DU penetrators @ 4 k each | |
| ft | ft | $ft^2$ | Btu | # DU | Btu W/# DU |
| 10 | 10 | 100 | 4.6 E+7 | 105.8 | 1.1 E+5 |
| m | m | $m^2$ | Cal | g DU | Cal W/g |
| 3.048 | 3.048 | 9.29 | 1.15 E+10 | 4.8 E+4 | 6 E+4 |

* Does not include the BTU values from the missile propellant, or the 3.8 l (1 gal) gasoline.

OTHER DU PENS TEST OXIDATION FACTORS

Other tests on DU pens of 105 mm missiles performed in a furnace (Elder and Tinkle 1980), related the degree of oxidization to air flow, temperature, and $CO_2$ content of the air stream, see Table B-3 data. High air flow and rapid temperature fluctuation, as well as high $CO_2$ contributed to more rapid oxidation. In 2 and 4 hr burns, the percent of oxidation was determined. The increased oxidation of uranium in 50% $CO_2$, at higher temperatures explains the reason $CO_2$ extinguishers are ineffective in uranium fires. The data in Table B-3 were used to reproduce the curves shown in Figure 3-4. Figure B-5 was reproduced by reading the data from Figures 22 and 23 in the test report (Elder and Tinkle 1980).

Table B-3 and Figures B-4 and B-5, show the effect of temperature, air flow, and $CO_2$ content on oxidation rate. The run N-1 (see bottom of Table B-3, left side), at 700 °C (1292 °F) shows (in Figure B-4), almost a 40% drop in mass loss when forced-air was stopped. The tests in Hilliard (1958), Table 1 and Figure 1, were completed at 500 cm/min air flow, whereas, the tests in Elder and Tinkle 1980, were completed in 223 cm/s air flow (blast furnace effect), at about 25 times greater flow. The rate of oxidation at these air flows is higher than the oxidation rate from the open burn given in the test by Hooker et al., 1983. The increase in available oxygen in a furnace, as well as the removal of combustion products by the convection in the furnace, accounts for much of the difference in burn time between open burns and furnace burns.

Table B-3. Oxidation of 105 mm Missile DU Penetrator. (Elder and Tinkle 1980)

| AIR AT 223 CM/S, 2 HR BURN TIME | | | CO, 50%, AIR 50% AT 223 CM/S | | | |
|---|---|---|---|---|---|---|
| Run # | Temp (°C) | Oxidation (%) | Run # | Burn Time (Hr) | Temp (°C) | Oxidation (%) |
| A-7 | 500 | 6.8 | M-1 | 2 | 500 | 6.0 |
| A-2 | 600 | 6.2 | M-2 | 2 | 600 | 6.3 |
| A-4 | 700 | 22.1 | M-3 | 4 | 700 | 21.3 |
| A-6 | 800 | 17.6 | M-4 | 4 | 800 | 29.9 |
| A-5 | 900 | 15.7 | M-5 | 4 | 800 | 30.2 |
| AIR AT ZERO VELOCITY - 2 HRS | | | M-6 | 4 | 900 | 24.9 |
| N-1 | 700 | 13.3 | M-7 | 4 | 1000 | 23.6 |

The Elder and Tinkle (1980) report mentions that some of the data was omitted because of a temperature fluctuation in the runs causing an increase in oxidation rate. This is consistent with statements in other documents where a fluctuation in temperature seemed to affect the oxide layer, allowing more rapid oxidation. The respirable fraction shown in Figure B-5, also shows a definite variation with temperature.

## TOTAL URANIUM OXIDATION CONSERVATISM

The uranium billet and fuel assembly total oxidation has been estimated based on extrapolated total oxidation time, fire duration, and projected fire cool down and this uranium oxidation is the basis for the fire event dose consequences. This total uranium oxidation is considered conservative for the following reasons:

### Extrapolated Billet Oxidation Time

The extrapolated average uranium billet total oxidation time based on information from smaller uranium-piece oxidation studies was determined to be about 177 hours. A value of 100 hours, nearly a factor of 2 higher oxidation rate, was determined to be a conservative total oxidation time for all of the uranium in the 3712 Building.

### 3712 Building Fire Temperature

The extrapolated average billet total oxidation time was based on the maximum fire temperature, 1093.3 °C (2000°F), associated with the maximum free combustible loading in the building. The average free combustible loading in the building yields a maximum fire temperature of 1031.1 °C (1888 °F).

### Estimated Cool Down Time

The cool down to below 300 °C (572 °F) of 8 hours, and the uranium oxidation time for a 4 hour free combustible fire, does not take into account

the differences in ash content between the DU pens fire tests and the 3712 Building fire. The lower ash content associated with 3712 Building fire would decrease the cool down time by about a factor of about 130, thereby reducing the amount of oxidized uranium.

Fuel Assembly Zircaloy-2 Cladding

Rather than develop an extrapolated total oxidation time for the fuel assemblies with there smaller weight to surface ratio than the billets, the extrapolated oxidation time for billets was used on the basis that the assemblies cladding provides some protection of the uranium. In addition, the cladding would have to oxidize off first whereas the dose consequences assume that all the oxidation is uranium oxidation.

The oxide on Zircaloy-2 is very tenacious, similar to that of aluminum. This is because of the crystalline structure and there is less change in volume when going from Zr to $ZrO_2$ as compared to the volume change in going from U to $U_3O_8$ (uranium is denser and the O to U atom ratio is greater). Because of this difference, the uranium oxide layer is exfoliated similar to the oxidation or rusting of iron. The difference in oxidation rates between uranium and Zircaloy-2, is supported by the cited Zircaloy ignition test where flames were applied directly to the Zircaloy tubing without impact. Therefore, in reality but without empirical information to support it, the cladding would be expected to provide considerable protection to the uranium in the assemblies that represents approximately 35% of the uranium in the 3712 Building.

Inventory

The oxidized uranium analysis is based on an inventory, current at the time of the Radiological Consequence Code (GENII) analysis, of 1122 MTU. Since the analysis, the uranium inventory in the 3712 Building has decreased to below 900 MTU. Impending uranium transactions may realize further inventory reduction.

## 3.0 IGNITION

The Baker (1966) study, Table 1 and Figure 10, shows a relationship between surface area/weight in $cm^2/g$ and ignition temperature. The test pieces in the study were all quite small compared to the uranium billets, where 13 out of 20 had one dimension less than 0.16 cm (1/16 in). The data from Table 1 of the study, are plotted in Figure B-6. A billet and a DU pen was added using estimated temperatures to fit the curve. Data from Hilliard (1985), shown on Figure B-6, are from tests at various temperatures; however, it is not stated that the test pieces ignited. The test pieces from Baker 1966, were predominately in the pyrophoric size range, with one dimension being 0.16 cm (1/16 in) or less. Extrapolation of this pyrophoric material ignition temperature information to the size of uranium billets gives an ignition temperature of 875 °C (1607 °F). This is below the estimated 3712 Building maximum combustible fire temperature of 1093 °C (2000 °F); however, the 3712 Building fire temperature is below the melting point of uranium 1132 °C (2070 °F). As to whether the uranium is actually considered ignited between 875 °C (1607 °F) and 1093 °C (2000 °F) is unknown. The results from

the testing of the DU pens (Mishima, et al., 1985), led the authors to the conclusion that the Du pens ignited at above 700 °C (1292 °C), if they can be ignited at all. The tests do indicate that the DU pens were intact after the test, the oxidation was not complete, and the oxidation was not sustained and this was with a fire test temperature and duration exceeding that for the projected 3712 Building fire, and for uranium pieces much smaller than those in the 3712 Building. There was no report of the DU pens having melted even though the fire temperature was well above the melting point; this may mean that the uranium oxidation layer or the fire ash may offer considerable protection to the uranium. Note that the oxidation rates for the billets developed in the previous section are based on empirical information at the estimated 3712 Building maximum combustible fire temperature of 1093 °C (2000 °F).

## 4.0 ZIRCALOY-2 CLAD FUEL ASSEMBLIES

The fuel assemblies are beta-quenched uranium, clad with Zircaloy-2. In the Cooper (1985) study, it is pointed out that finely divided zirconium and Zircaloy once ignited burn rapidly even under water. Larger pieces of Zircaloy are harder to ignite. In tests, heating Zircaloy tubing with burning Zircaloy powder and a torch did not ignite the tubes, Table B-4. In seven tests, the tubes 8 mm in length by 15 mm in diameter, and a wall thickness of 0.8 mm were heated by igniting Zircaloy-2 powder. Two of the tests were conducted using a torch. A tough oxide layer formed on the Zircaloy tubes, but ignition did not take place. The oxide that forms on zirconium and Zircaloy is much tougher than the oxide layer that forms on uranium. The metallurgical bond formed between the Zircaloy-2 cladding and the uranium when the fuel elements are co-extruded [temperature greater than 600 °C (1112 °F)] causes the Zircaloy-2 and uranium to expand and contract together. The Zircaloy-2 cladding increases the difficulty of oxygen penetrating through to the uranium. The fuel assemblies are much more resistent to oxidation than the uranium billets, if they can be made to oxidize at all. Since the Zircaloy-2 tubes did not ignite, it can be implied that the fuel assemblies will not ignite; however, to say that no oxidation of the fuel assemblies would take place in a prolonged fire would require further research.

Table B-4. Ignition Tests Performed on Zircaloy-2 Tubing Sections. (Cooper 1985)

| TEST | POWDER MASS (g) | POWDER TO TUBE RATIO (g/g) | POWDER IGNITION TEMPERATURE (°C) | POWDER BURN TIME (s) | TEMPERATURE (°C) | REMARKS |
|---|---|---|---|---|---|---|
| 1 | 0.375 | 0.25 | 430 | 30 | 822 | Tubing did not ignite; all tubes had an oxidized surface after the test. Tubing dimensions 8 mm by 15 mm diam. except test 5 & 6 which used half ring sections, corresponding weights 1.5 & .75 g. |
| 2 | 1.5 | 1 | 430 | 80 | 585 | |
| 3 | 3.0 | 2 | 430 | 50 | 941 | |
| 4 | 6.0 | 4 | 430 | 70 | 894 | |
| 5 | 6.0 | 8 | 430 | 25 | 1150 | |
| 6 | 6.0 | 8 | 400 | 60 | 798 | |
| 7 | 6.0 | 4 | 400 | 50 | 1098 | |
| 8 | 6.0 | 4 | 400 | 65 | 1600 | Powder + torch used in attempt to ignite tubing. Tubing glowed but ceased with torch removal. |
| 9 | 6.0 | 4 | 400 | 70 | 1600 | |

Ignition Powder: 300 mesh (50 μm) Zircaloy-2 powder (tests 1-5); 10-12 μm zirconium sponge powder(tests 6-9).
Heat source: Gas Torch, + Oxygen torch for test 9.
Starting conditions: Materials at ambient conditions on a ceramic base.

The oxide layers on uranium resist oxidation by preventing the oxygen from reaching the uranium metal. One of the means the oxide layer is broken is the expansion and contraction of the metal; therefore, allowing oxygen to reach the metal. Since the Zircaloy-2 cladding is more resistant to oxidation there would be less access to the uranium metal for the oxygen. Adding the resistance of the Zircaloy-2 and uranium oxide layers to prevent oxidation, the reduced expansion of the uranium/Zircaloy-2 bond, gives a combination that is highly resistant to oxidation. Given the additional resistance to oxidation of the fuel assemblies because of the cladding, assuming an oxidation time for the fuel assemblies which have a smaller weight to area ratio than the billets equal to that for the uranium billets, is reasonable.

## 5.0 CONCLUSIONS

The uranium in the 3712 Building will not ignite, but will oxidize under high temperature conditions. The extrapolated average uranium billet total oxidation time based on information from smaller uranium-piece oxidation studies was determined to be about 177 hours. A value of 100 hours, nearly a factor of 2 higher oxidation rate, was determined to be a conservative total oxidation time for all of the uranium in the 3712 Building.

## 6.0 REFERENCES

Baker, L. Jr., J. G. Schnizlein, and J. D. Bingle, 1966, "The Ignition of Uranium," *Journal of Nuclear Materials*, No. 20, pp. 22-38, North-Holland Publishing Company, Amsterdam New York.

Cooper, T. D., 1985, *Review of Zirconium-Zircaloy Pyrophoricity*, RHO-RE-ST-31 P, Rockwell International, Richland, Washington.

Elder, J. C., and M. C. Tinkle, 1980, *Oxidation of Depleted Uranium Penetrators and Aerosol Dispersal at High Temperatures*, LA-8610-MS, University of California, Los Alamos Scientific Laboratory, P. O. Box 1663 Los Alamos, New Mexico.

Hilliard, R. K., 1958, *Oxidation of Uranium in Air at High Temperatures*, HW-58022, General Electric, Hanford Atomic Products Operation, Richland Washington.

Hooker, C. D., D. F. Hadlock, J. Mishima, and R. L. Gilchrist, 1983, *Hazard Classification Test of The Cartridge, 120 mm, APFSD-T, XM829*, PNL-4459, Pacific Northwest Laboratory, Richland, Washington.

Mishima, J., M. A. Parkhurst, R. I. Scherpletz, and D. E. Hadlock, 1985, *Potential Behavior of Depleted Uranium Penetrators under Shipping and Bulk Storage Accident Conditions*, PNL-5415-UC-41, Pacific Northwest Laboratory, Richland, Washington.

Myott, C. F., 1993, *Fire Loading Calculations for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-TI-053, Rev 0, Westinghouse Hanford Company, Richland, Washington.

Figure B-1. $W_o/A_o$ in mg/cm$^2$ Related to Combustion Time.

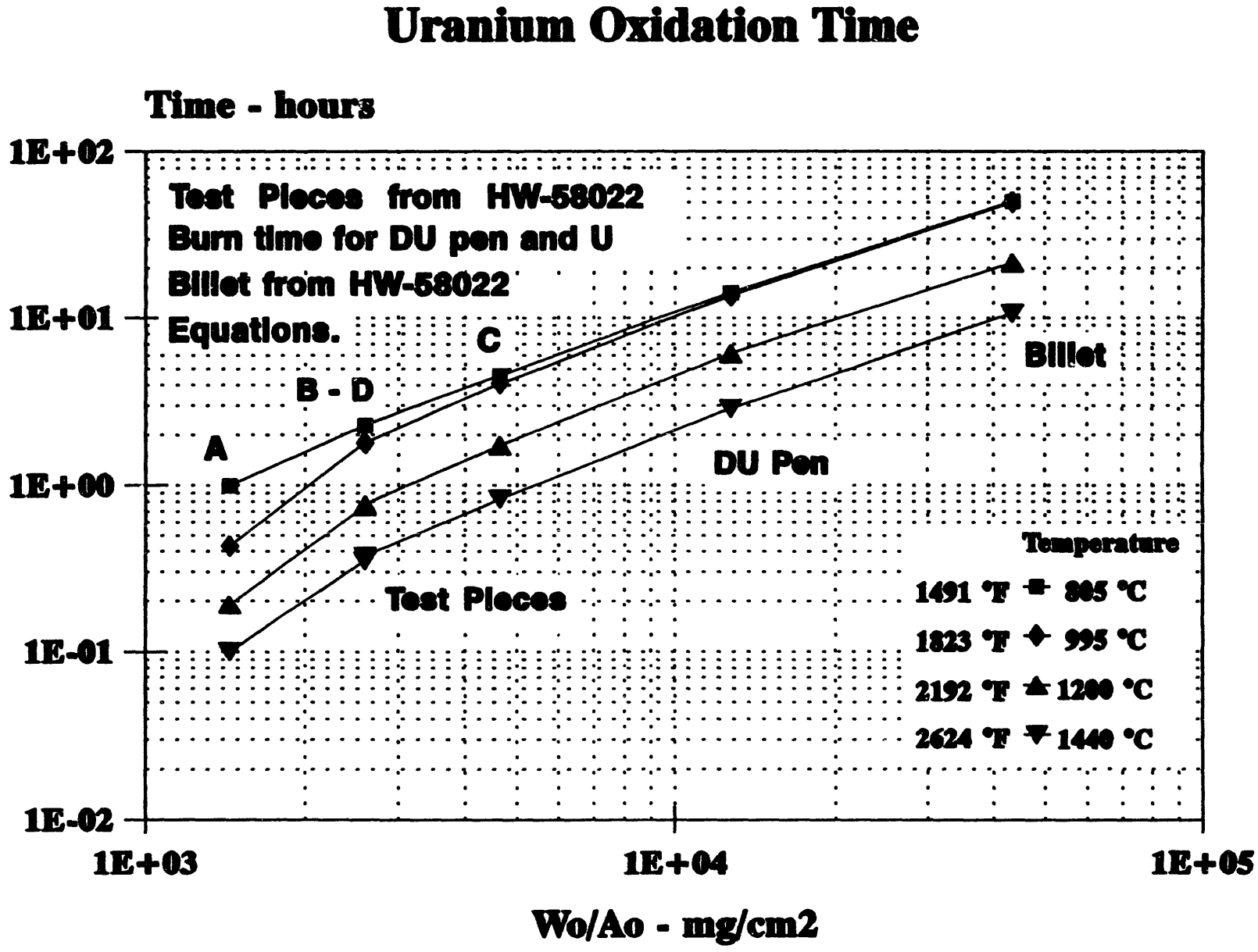


Figure B-2. Self-Heating Curve Furnace Temperature Under 400 °C (752 °F).

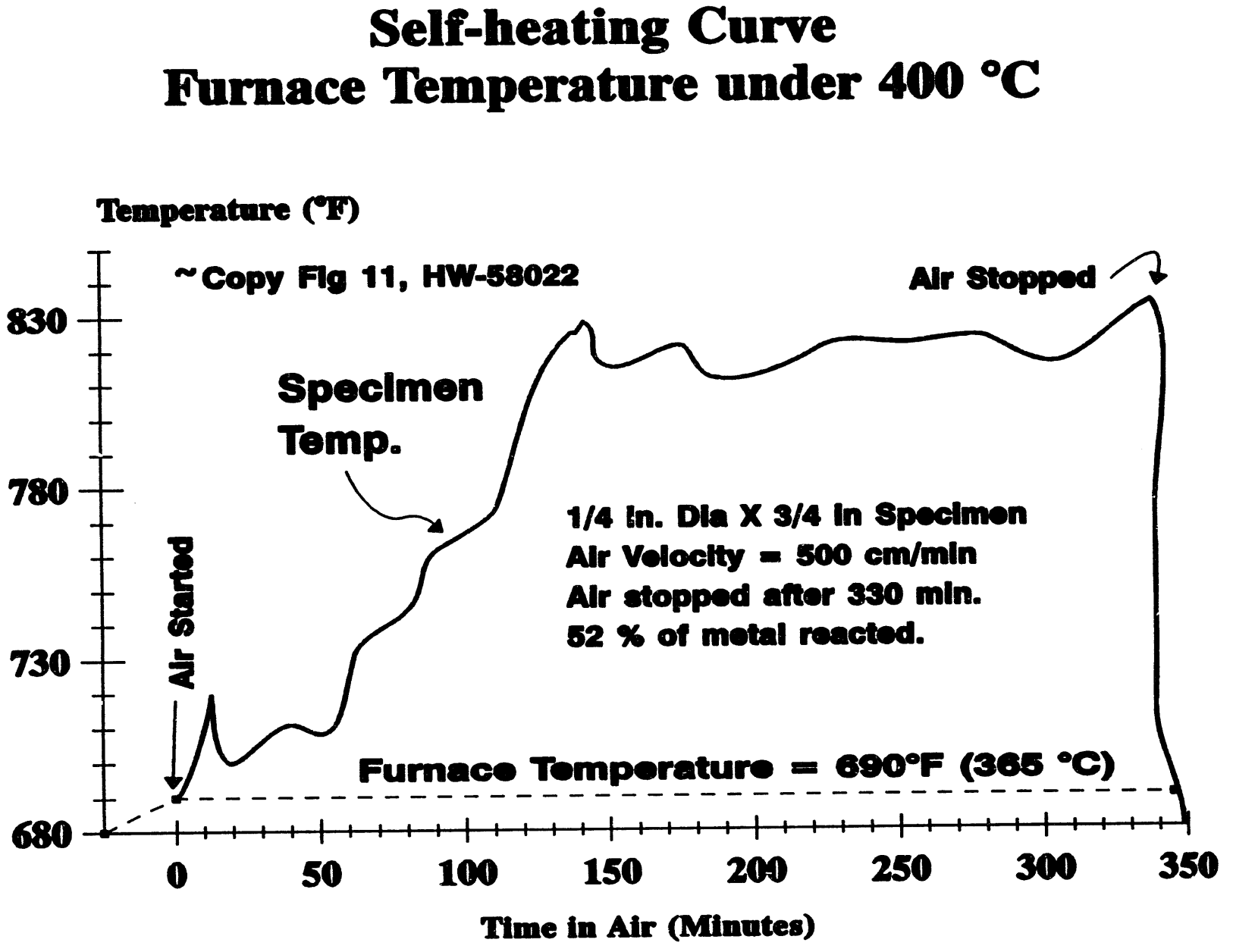


Figure B-3. Self-Heating Curve Furnace Temperature Above 700 °C (1292 °F).

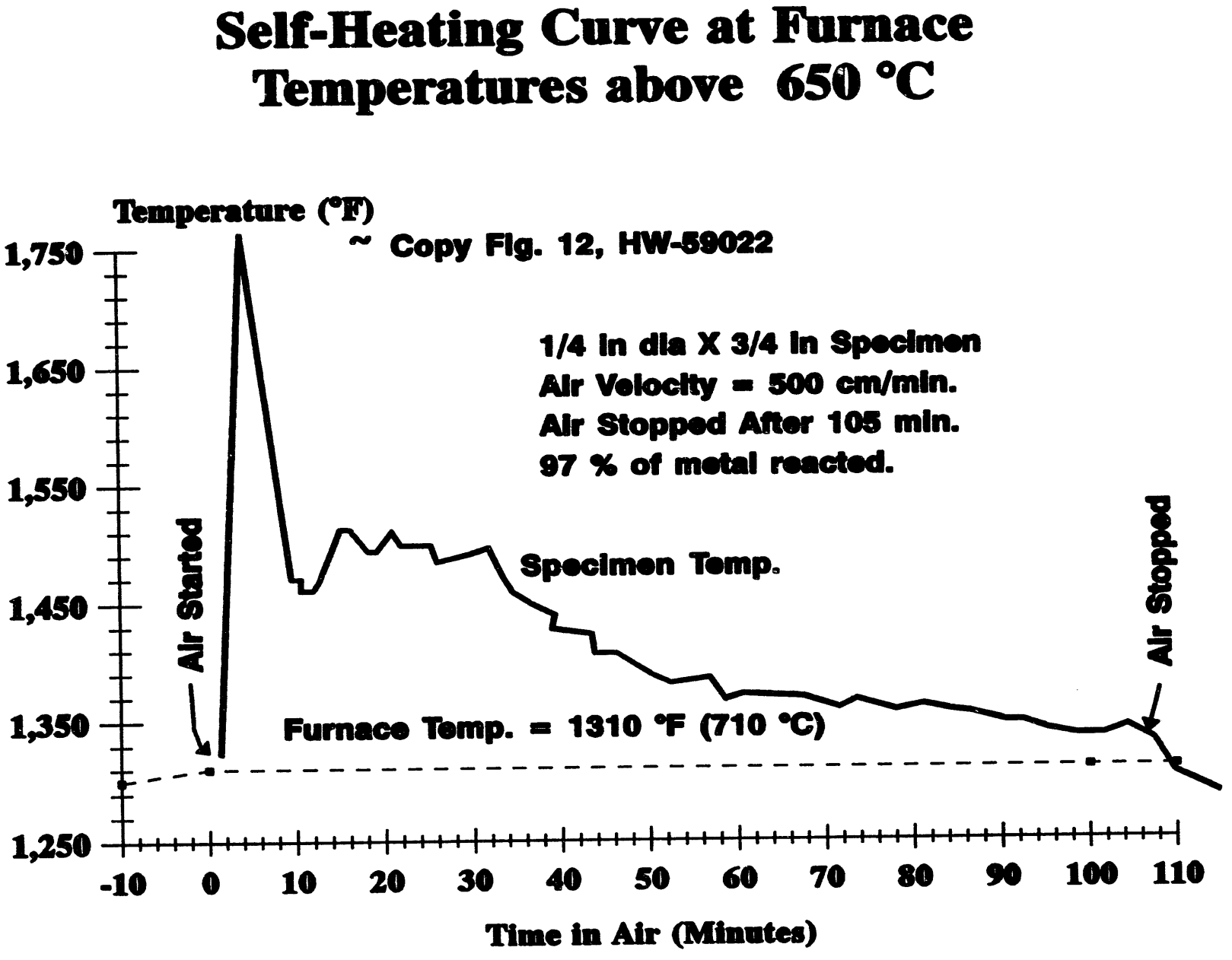


Figure B-4. Oxidation as a Function of Temperature.

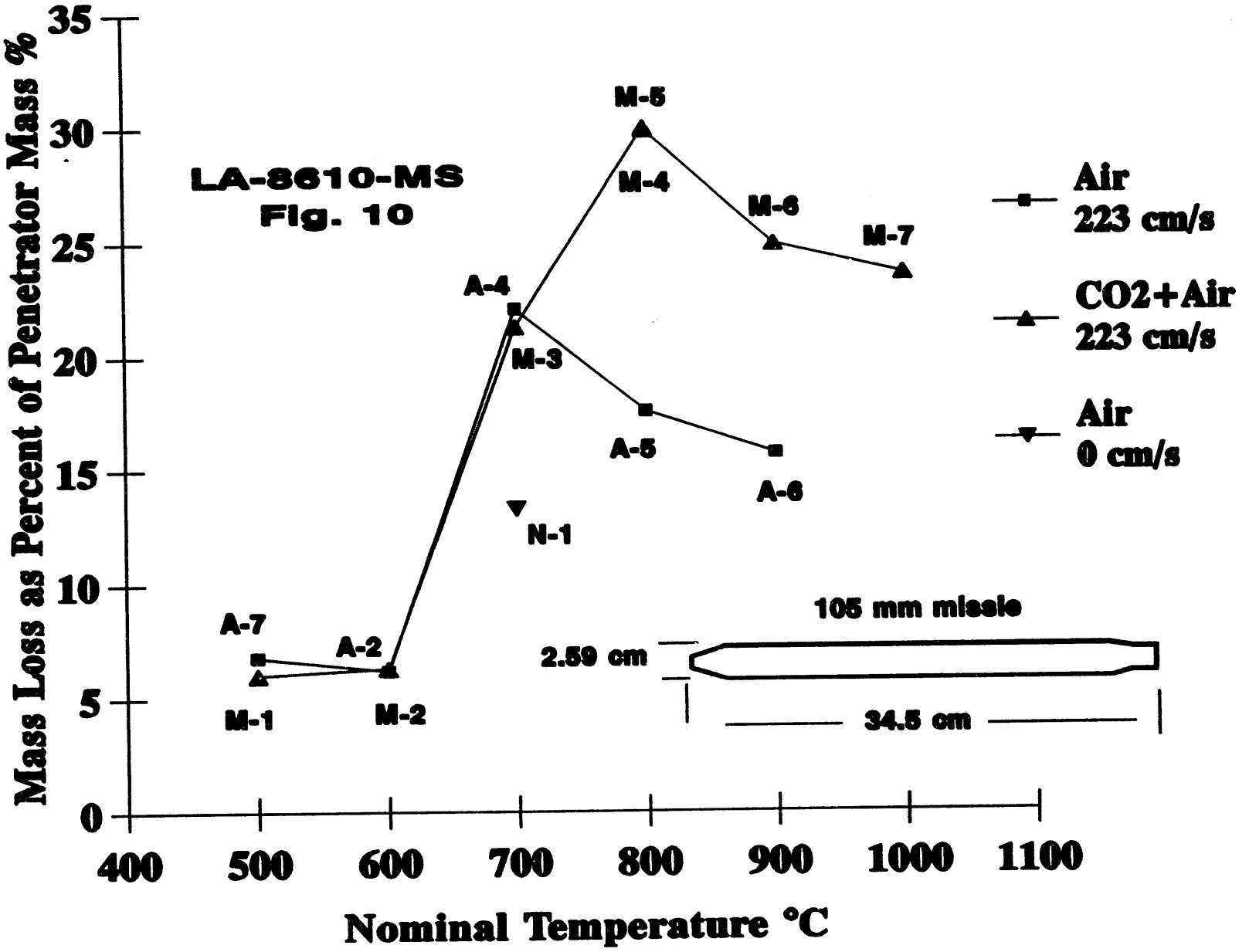
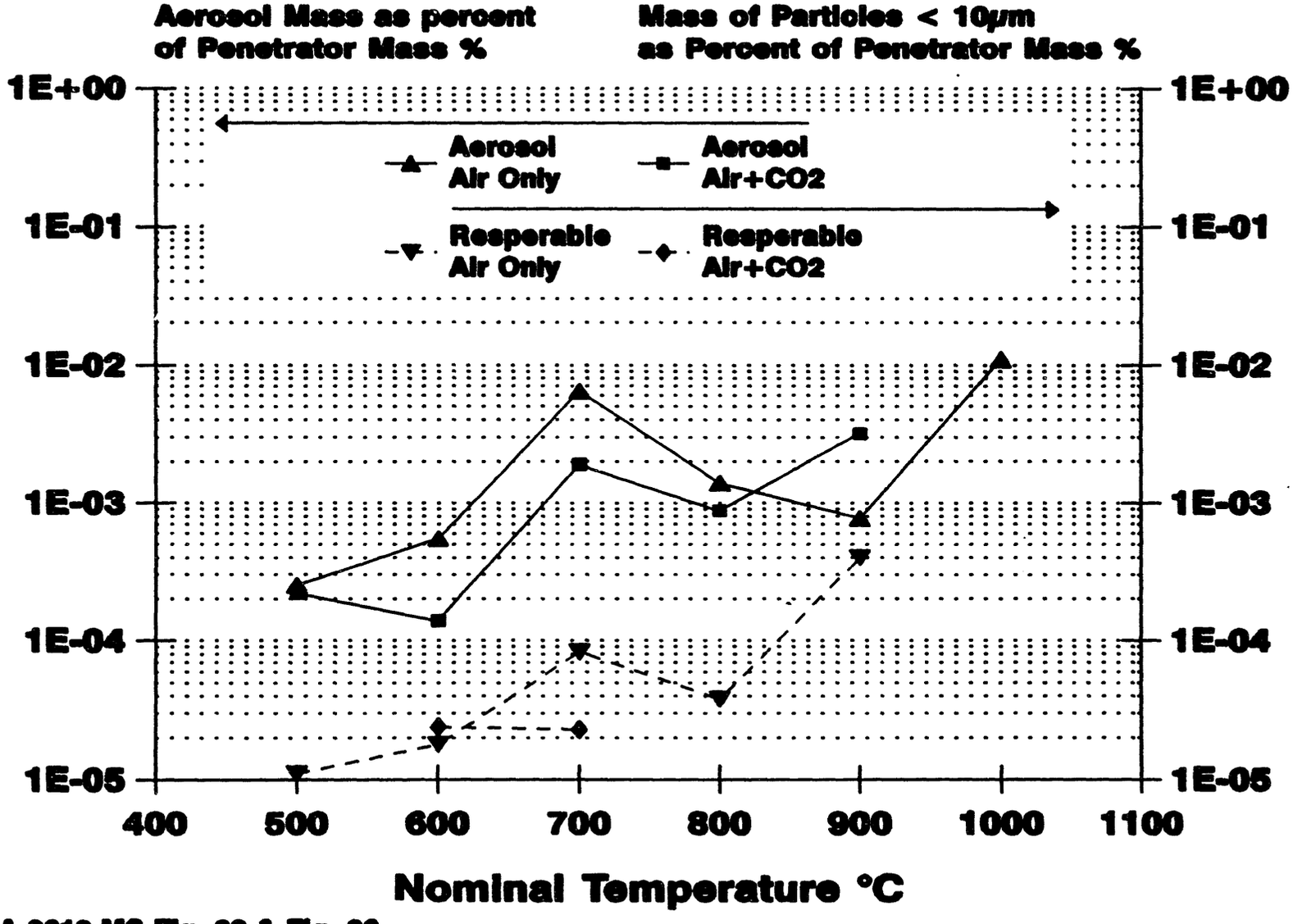


Figure B-5. Mass Loss as a Function of Temperature.




LA-9610-MS Fig. 22 & Fig. 23

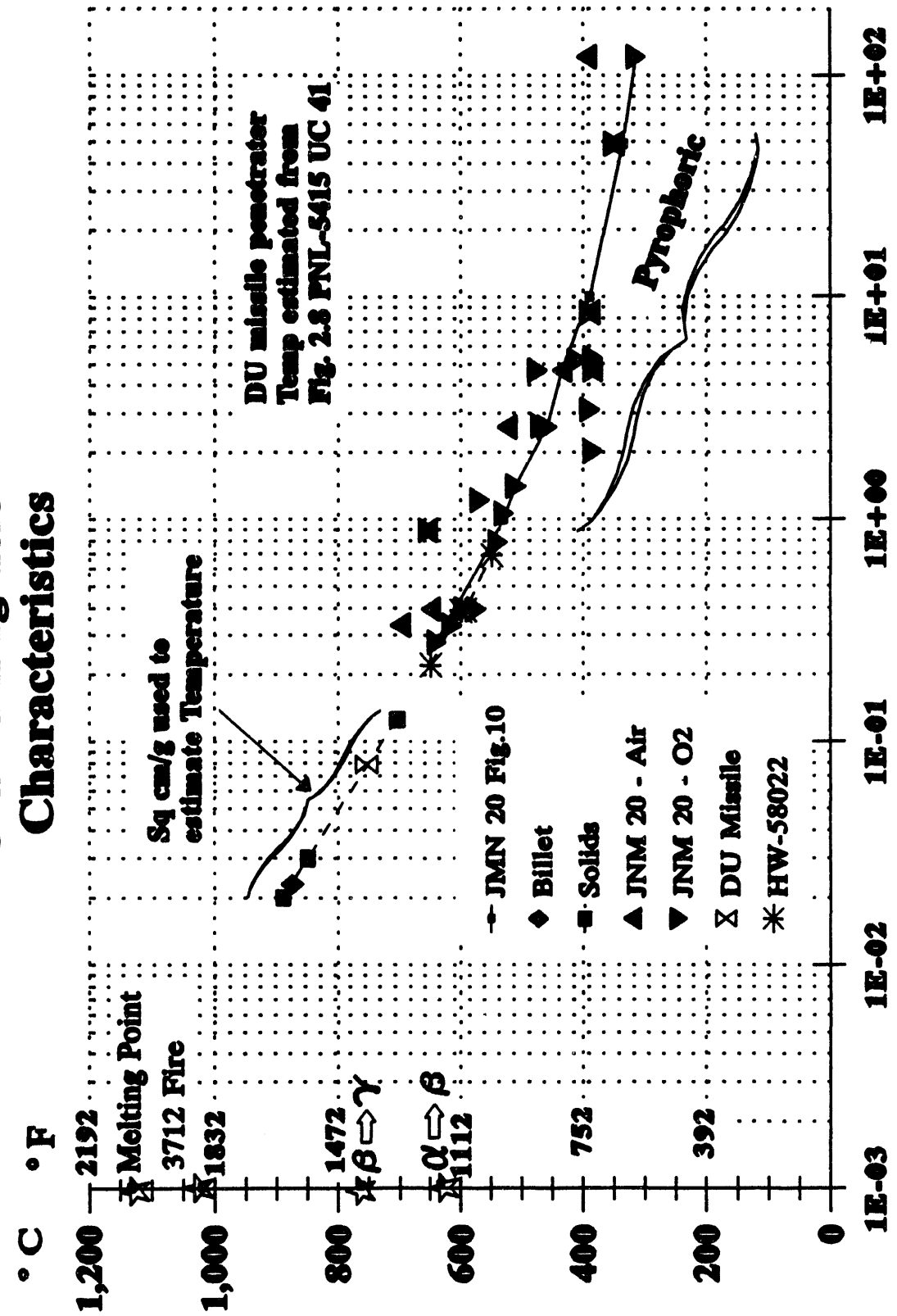
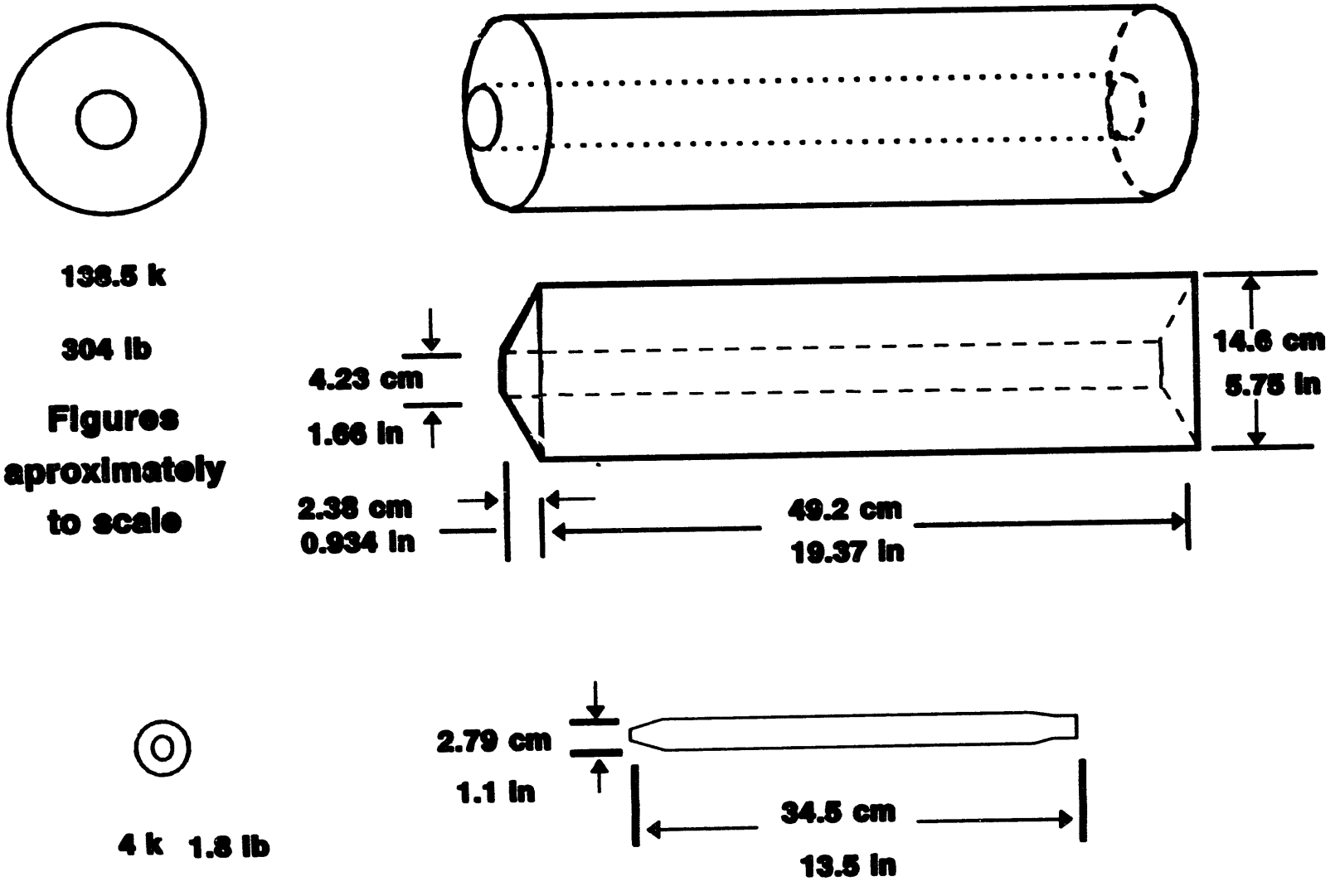



Figure B-6. Uranium Ignition Characteristics.

Figure B-7. Average Billet and Penetrator.



This page intentionally left blank.

DATE
FILMED
9/22/94

END