ENGINEERING CHANGE NOTICE

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Page 1 of 26

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2. ECN Category (mark one) Supplemental ☐ Direct Revision ☒ Change ECN ☐ Temporary ☐ Standby ☐ Supersedeure ☐ Cancel/Void ☐	3. Originator's Name, Organization, MSIN, and Telephone No. Terry S. Vail, Nuclear Safety, T4-56, 373-2092	4. USQ Required? X Yes KN No	5. Date 03/29/99
12a. Modification Work [] Yes (fill out Blk. 12b) [X] No (NA Blks. 12b, 12c, 12d)	12b. Work Package No. N/A	12c. Modification Work Complete N/A Design Authority/Cog. Engineer Signature & Date	12d. Restored to Original Condition (Temp. or Standby ECN only) N/A Design Authority/Cog. Engineer Signature & Date
13a. Description of Change The references were changed to reflect the Project Hanford Management Contract (PHMC) and the Waste Management Federal Services of Hanford, Inc. (WMH) procedures. The text was modified to reflect configuration changes in preparation of the deactivation of the 340 Vault tanks and the termination of the railcar loadout. Appendix A was replaced with the hydrogen generation analysis, Appendix B-D remain unchanged and Appendix E was added with the DOE-RL approval of revision 1 of the ISB.			
14a. Justification (mark one) Criteria Change ☐ Design Improvement ☐ Environmental ☐ Facility Deactivation ☐ As-Found ☒ Facilitate Const ☐ Const. Error/Omission ☐ Design Error/Omission ☐			
14b. Justification Details Annual reviews and any needed updates are required by HNF-PRO-700, "Safety Analysis and Technical Safety Requirements". The implementation of the new PHMC procedures and the WMH procedures provided enough modification of text and references to warrant a full revision. Also, the deactivation of the process of receiving radioactive liquid waste is sufficient cause for a complete revision.			
15. Distribution (include name, MSIN, and no. of copies) See Distribution page.			

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3 of 6

UNREVIEWED SAFETY QUESTION (USQ) SCREENING AND EVALUATION

1. Identification Number: 99-006

USQ SCREENING

Page 1 of 2

2. Title: 340 Waste Handling Facility Interim Safety Basis, Revision 2

INSTRUCTIONS: Respond to each question and provide justification for each response. Restatement of the question does not constitute satisfactory justification or basis. An adequate justification provides sufficient explanation such that an independent reviewer could reach the same conclusion based on the information provided [DOE 5480.21, 10.e.1].

QUESTIONS:

1. Does the proposed change or occurrence represent a change to the facility or procedures as described in the Authorization Basis?

☐ N/A ☐ No ☒ Yes/Maybe

Basis: The primary changes to the ISB include updating the references from the Westinghouse Hanford Company (WHC) controlled manuals to the Project Hanford Management Contract (PHMC) and Waste Management Federal Services of Hanford, Inc. (WMH) procedures. Also, references to cancelled or outdated documents were deleted and some of the text was made consistent with the new procedures. Many text changes were made to reflect minor physical and operational changes that have occurred; and some of the more significant of these follow. The fissionable material inventory, discussed in section 1.6, was reworded to be less than 1/3 of a minimum critical mass, consistent with HNF-PRO-334s' definition of an Isolated Facility. Section 1.6 was changed to identify the possibility that waste may be pumped from the 340 vault tanks to a waste tanker instead of a railcar. Section 2.2 was changed to clearly indicate that existing in-line valves in the underground piping from the 324, 325, 326, 327 and 329 Buildings to the vault tanks are closed and the pipes will be cut and capped. Section 2.2, 2.3.1 and 2.3.3 were reworded to indicate that containerized waste can no longer be discharged into the vault tanks through the sump in the 340 decontamination area or the lines in the 340-B load out facility. However, if decontamination activities require the use of the piping the valves can be opened. Text was added to Section 2.2 that indicates the deactivation and isolation of the RLWS transfer piping negate the need for maintenance of the valve boxes and associated instrumentation i.e. sump leak detectors. Wording was added to Section 2.3 that state deactivation activities will be governed by HNF-IP-1000 that is derived from HNF-PRO-072. Also, if reactivation is necessary relevant precautions are exercised. Section 2.3.1 addresses the switch from panel mounted to computerized annunciator. Text was added to Section 3.1.1 to indicate that an automatic sprinkler system was added to 340-B east and west. Words were added to Section 3.1.1.1 that indicate a spray release accident for the load-out of a volume of waste equal to or smaller than the railcar would be bounded by the spray release accident considered for hazard category. Also in 3.1.1.1, the fissile inventory for AGS tanks (and associated hazard category calculations) were revised to reflect recent clean-out impact. Section 3.1.3.1 was reworded to indicate that 340 and 340-B have fire detection and suppression capabilities. Section 3.1.3.1 was revised to describe the "proposed" natural gas fired steam boilers as now a reality. [The risk of using the underground natural gas pipeline and the use of the gas was already determined under Rev 1-B to be acceptable.] Section 3.1.3.2, the discussion of the sump in 340-A draining to the vault tanks was deleted. Text in Section 3.1.3.3 was added that provides the conclusion that a helicopter rescue service does not change the conclusion that a helicopter or plane crash would produce minimal airborne release and is incredible. Section 3.1.3.7 added a discussion on hydrogen generation through radiolysis. Section 3.1.4.1 text was added that indicated the pumping of any volume of waste less than 57 kL with a composition bounded by the source term can not exceed the dose-consequence calculations. Also, the release of chemicals can not exceed the ERPG. Section 3.2.1.1, text was added that indicate that some ventilation alarms announce locally at the stack and some announce remotely through the alarm horns in the 340 control room and through the 310 TEDF plant alarms. Section 3.2.1.1 the description of the process radiation detector and alarms in Valve Box 9 was deleted because of the RLWS isolation. Also, the text about the liquid level alarms in the RLW lines at 325 and 327 Buildings was deleted. Section 3.2.1.2 the discussion of the area radiation monitor and alarms located inside the 340-A west door is deleted. Section 3.2.1.3 was reworded to indicate the smoke detectors and alarms at central dispatch were deleted and the automatic heat activated sprinklers were added at 340-B. Section 3.2.2.3.3 text was reworded to identify the loadout of waste to a waste container that is not specific to the railcar. Section 3.2.3.4.1 text was rewritten to reflect the new PHMC policy. Section 4.4 the text was reworded to reflect current PHMC policy. Deleted the Conduct of Operations Comparative Evaluation (App. A), added the hydrogen generation calculations as new App. A, and appended the current SER as appendix E.

2. Does the proposed change or occurrence represent conditions that have not been analyzed in the Authorization Basis?

☐ N/A ☐ No ☒ Yes/Maybe

Basis: The text was revised to include the facility configuration changes that terminated the capability to receive and store waste through the Radioactive Liquid Waste System (RLWS). The text in section 1.2 was changed to indicate that if liquids in the Retention Process Sewer (RPS) become radioactive that upstream facilities would divert the flow to a segregated basin. Also, some of the facilities may retain the flow within the facility. Section 1.4 was changed to indicate that as of October 1, 1998, operation of the RLWS was terminated. A footnote was added in section 1.6 that indicates the consequences of closure of the RLWS to new waste receipts result in a residual inventory which may only increase slightly due to decontamination activities so the source term used in the previous hazard evaluation is considered conservative. Section 2.2 was worded to allow the vault tanks to receive liquids as a result of decontamination activities. A description of the

UNREVIEWED SAFETY QUESTION (USQ) SCREENING AND EVALUATION

1. Identification Number: 99-006

USQ SCREENING

Page 2 of 2

2. Title: 340 Waste Handling Facility Interim Safety Basis, Revision 2

computerized monitoring and alarming capabilities was inserted into Section 2.3.1. Text in Section 2.3.3 was added which indicate that a different waste tanker will require new adapters for the vent and fill lines. The fissile material inventory in the 340-A above ground tanks was adjusted from 34 to 46 grams to the conservative estimate of < 11 grams in Section 2.4.3. The calculations in Section 3.1.1 were redone with the 11 grams of fissile material, concluding the facility is still a Hazard Category 3. Section 3.1.3.6 the discussion about the loss of electricity with transfer lines full of liquid was reworded to indicate that the isolation of the lines does not allow the lines to be full of liquid. Section 3.1.3.7, the discussion about the use of high radiation monitors was deleted. Section 3.1.3.7, the discussion providing conclusions about hydrogen generation is added with the conclusion that additional controls are not needed. Section 3.2.2.2.2 was reworded to provide consistency with PHMC and WMH procedures. Section 3.2.2.3.1 text was deleted that provided for the separate facility classification of the solid waste system due to low quantities of fissile material. Section 3.2.3.4.2 some text was deleted which identified stationary air-sampling instrumentation on the decon stack and the vault and valve pits. While most of these changes do not warrant any special analysis (since they either are themselves covered under broad existing evaluations, or are ameliorated by the shutdown impacts) the fact remains that some have not been formally analyzed. The subsequent USQ Evaluation should explore this in more detail.

3. Does the proposed change represent a test or experiment NOT described in the Authorization Basis that may affect the safe operation of the facility?
☐ N/A ☒ No ☐ Yes/Maybe

Basis: This change does not constitute a test or experiment.

4. Does the proposed change or occurrence represent a change to the Technical Safety Requirements or a reduction in the margin of safety defined in the Technical Safety Requirements?

☐ N/A ☒ No ☐ Yes/Maybe

Basis: There are three IOSRs for the 340 Facility; 1) maximum fissile material limit, 2) solid waste storage limit, and 3) 340-B Waste Loadout Restrictions. None of the changes result in a reduction of the margin of safety established by the IOSRs. The isolation of the RLWS and termination of the receipt and storage of radioactive liquid waste establishes an overall reduction of hazardous material at the 340 facility.

USQE #1 R.T. Stordure

(Print Name)

USQE #2 D.L. Halgren

(Print Name)

(Signature)

(Date)

(Signature)

(Date)

The following guidance should be considered when completing this screening. This guidance should not be considered all-inclusive; additional factors may need to be considered, depending on the nature of the proposed change.

Does the proposed change:

- 1) Modify, add or delete a safety class function of a structure, system or component stated in the authorization basis?
- 2) Alter the design of a structure, system or component as described in the authorization basis?
- 3) Modify, add or delete the description of operation, operating environment, or analyses of any system or component described in the authorization basis?
- 4) Modify, add, delete or conflict with any of the design bases stated in the authorization basis?
- 5) Conflict with the principle or general design criteria stated in the authorization basis?
- 6) Modify, add or delete any plant design features described in the authorization basis?
- 7) Modify, add or delete a flow diagram or facility drawing provided in the authorization basis?
- 8) Create the potential for new system or component interactions (e.g., seismic, electrical breaker coordination)?

UNREVIEWED SAFETY QUESTION (USQ) SCREENING AND EVALUATION

1. Identification Number: 99-006

USQ EVALUATION

Page 1 of 2

2. Title: 340 Waste Handling Facility Interim Safety Basis, Revision 2

1. Does the PROPOSED CHANGE, test, experiment or DISCOVERY increase the probability of occurrence of an accident previously evaluated in the AUTHORIZATION BASIS documentation?
☒ No ☐ Yes/Maybe

BASIS: The probability of occurrence of the evaluated accidents is a qualitative application of the annual frequency category descriptions provided in ISB Table 3-5. By demonstrating that the 340 Facility can be operated safely, assuming worst-case unmitigated releases from the highest severity/highest probability accidents, it can be concluded that the facility can also be operated safely assuming the unmitigated accident releases of the lower probability/lower severity accidents. The worst-case accidents having the highest severity/highest probability that were selected for evaluation include the spray release upon failure of the pipe or pipe connection while pumping liquid into a railroad tank car and a fire occurs in the 340-B West dry-waste storage area. The probability of a fire in the 340 B West building remains unchanged, because no changes affecting the occurrence of a fire were made. The railroad tank car can no longer be loaded with radioactive waste pumped from the vault tanks, so there is zero probability of that accident per se, although a spray release could still occur during the filling of a highway tanker, which the ISB now accommodates. The probability of a spray release should be reduced significantly because of the isolation and deactivation of the RLWS. However, there's some trade-off involved, as the number of transfers needed to ship a given volume of residual waste (or future clean-up or RPS off-spec waste) could increase due to the loss of the 20,000 gal rail tankers. [also, separate USQs have already dealt with individual changes such as a new medical helicopter service, isolating RLWS, annunciator panel, etc.]

2. Does the PROPOSED CHANGE, test, experiment or DISCOVERY increase the consequences of an accident previously evaluated in the AUTHORIZATION BASIS documentation?
☒ No ☐ Yes/Maybe

BASIS: The consequences or severity of evaluated accidents are not increased by the changes. The severity of the evaluated hazards is identified in Table 3-4. The accidents with the highest severity (Category II) include 1) a LPG tank BLEVE, 2) a helicopter or plane crash, 3) a naturally occurring (seismic) event with a railcar in 340-B East, and 4) a criticality. Since the 340 Facility will not receive any more radioactive liquid waste, the overall source term will not increase above the residual heels present in the system, which is lower than the source term used for the hazard category. Since any substitute for the rail tank car will involve smaller amounts of radioactive liquid waste from the 340 Facility, the consequences of an accident should be reduced. The fissile material inventory limits established for an Isolated Facility are not changed and are less than a minimum critical mass, and while the probability of a criticality is further reduced, the consequences of a criticality evaluated in the preliminary hazard analysis Table 3-4 will remain unchanged. The radiological and toxicological consequences of a helicopter or plane crash will be reduced since the quantity of material available for release is reduced.

3. Does the PROPOSED CHANGE, test, experiment or DISCOVERY increase the probability of occurrence of a malfunction of EQUIPMENT IMPORTANT TO SAFETY (ITS EQUIPMENT) previously evaluated in the AUTHORIZATION BASIS documentation?
☒ No ☐ Yes/Maybe

BASIS: There are no pieces of ITS under the 340 A/B. However, some changes made to related-type systems are mentioned here only to address the spirit of this question. The deactivation and isolation of the RLWS negates the need for the process radiation detector located in valve box 9, the area radiation monitor in 340-AA, and the low-point alarms at 325 and 327. Eliminating these devices effectively reduces the probability of a malfunction.

4. Does the PROPOSED CHANGE, test, experiment or DISCOVERY increase the consequences of a malfunction of ITS equipment previously evaluated in the AUTHORIZATION BASIS documentation?
☒ No ☐ Yes/Maybe

BASIS: As noted there are no ITS items at 340. The hazards detected and monitored by related equipment described under 3 (above) are reduced with the termination of the receipt and storage of liquid radioactive waste. The residual heels of radioactive material available for release, in the vault and above ground tanks, is smaller than the quantities evaluated for the hazard category. The reduced volume of material and reduced operations result in a reduced consequence of loss of that equipment.

6 of 6

UNREVIEWED SAFETY QUESTION (USQ) SCREENING AND EVALUATION

1. Identification Number: 99-006

USQ EVALUATION

Page 2 of 2

2. Title: 340 Waste Handling Facility Interim Safety Basis, Revision 2

5. Does the PROPOSED CHANGE, test, experiment or DISCOVERY create the possibility of an accident of a different type than any previously evaluated in the AUTHORIZATION BASIS documentation?
☒ No ☐ Yes/Maybe

BASIS: The change does not create the possibility of an accident different than previously evaluated. The evaluation of a hydrogen deflagration was not provided in the original ISB, however the possibility of this type of accident is considered low. The recommendation is that adequate controls assuring safe hydrogen in air concentration are maintained and additional controls are not needed. Also, the curie content of the waste received in the vault tanks being much less than the bounding source term supports the conclusion that the potential of occurrence is low. The evaluation provides quantitative support for the recommendations, but does not conclude the accident is possible. Other changes provide for a reduction of the number of radioactive material handling operations, but the procedures used for the limited number of handling operations remain unchanged.

6. Does the PROPOSED CHANGE, test, experiment or DISCOVERY create the possibility of a malfunction of ITS EQUIPMENT of a different type than any previously evaluated in the AUTHORIZATION BASIS documentation?
☒ No ☐ Yes/Maybe

BASIS: There are no ITS items at 340. The reduced number of operations involving the handling of radioactive material result in the reduced reliance on any safety related equipment. The changes to the ISB do not include operations or processes different from previously evaluated; therefore the possibility of malfunctioning equipment during operations remains unchanged.

7. Does the PROPOSED CHANGE, test, experiment or DISCOVERY reduce the margin of safety as defined in the basis for any Technical Safety Requirement?
☒ No ☐ Yes/Maybe

BASIS: The changes do not reduce the margin of safety as defined by the basis for the IOSRs. The three established to ensure compliance with the Hazard Category 3 facility include a limit on fissile material, the solid waste storage limit and 340-B railcar loadout restrictions. The changes to the ISB do not impact the fissile material limit or the solid waste storage limit. The deactivation and isolation of the RLWS terminate the process of loading waste into a railcar, but allows for the loadout of a waste container of the same volume or less. Therefore, the impact to the IOSRs is equal to or less than previously evaluated.

8. Does the PROPOSED CHANGE, test, experiment or DISCOVERY require a new or revised Technical Safety Requirement?
☒ No ☐ Yes/Maybe

BASIS: The changes to the 340 Facility ISB do not require new or revised IOSRs.

USQE #1: R.T. Stordeur
(Print Name)

USQE #2: D.L. Halgren
(Print Name)

Signature

Date: 3/31/99

Signature

Date: 3/31/99

PRC REVIEW (If Required)

Meeting No.: 99-1

Date: 3/31/99

PRC Chairman Concurrence:

Date: 3/31/99

See associated meeting minutes.

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340 Waste Handling Facility Interim Safety Basis ^S

Terry S. Vail

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Key Words: 340 Facility, safety analysis, preliminary hazards analysis, interim safety basis, 300 Area, storage tanks, operational safety requirements, risk assessment, USQ process.

Abstract: This document presents an interim safety basis for the 340 Waste Handling Facility classifying the 340 Facility as a Hazard Category 3 facility. The hazard analysis quantifies the operating safety envelop for this facility and demonstrates that the facility can be operated without a significant threat to onsite or offsite people.

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**340 WASTE HANDLING FACILITY INTERIM
SAFETY BASIS**

HNF-SD-WM-ISB-003

Rev. 2

March 1999

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CONTENTS

LIST OF FIGURES	iv
LIST OF TABLES	iv
LIST OF TERMS	v
1.0 INTRODUCTION AND SUMMARY	1-1
1.1 PURPOSE	1-1
1.2 BACKGROUND	1-1
1.3 FACILITY STATUS	1-2
1.4 FUTURE ACTIVITIES	1-2
1.5 CONFIGURATION MANAGEMENT	1-2
1.6 SUMMARY OF SAFETY ANALYSIS AND SUPPORTING INFORMATION	1-3
1.7 CONCLUSIONS	1-4
2.0 FACILITY DESCRIPTION	2-1
2.1 GENERAL	2-1
2.2 PROCESS/OPERATION	2-2
2.3 BUILDINGS	2-3
2.3.1 340 Building	2-3
2.3.2 340-A Building	2-4
2.3.3 340-B Building	2-4
2.3.4 MO-036 Building	2-5
2.3.5 MO-741 Building	2-5
2.4 RADIONUCLIDE AND HAZARDOUS MATERIAL INVENTORIES	2-5
2.4.1 340 Vault	2-5
2.4.2 340 Building	2-5
2.4.3 340-A Building	2-6
2.4.4 340-B Building	2-6
3.0 INTERIM SAFETY BASIS EVALUATION	3-1
3.1 HAZARD ANALYSIS	3-1
3.1.1 Hazard Categorization	3-1
3.1.2 Hazard Categorization Methodology	3-2
3.1.3 Preliminary Hazards Analysis	3-3
3.1.4 Accident Scenarios and Consequences	3-6
3.1.5 Risk Evaluation	3-7
3.2 HAZARD CONTROLS	3-8
3.2.1 Structures, Systems and Components	3-8
3.2.2 Administrative Controls/Operational Restrictions	3-10
3.2.3 Institutional Safety Programs	3-14
3.3 SUMMARY	3-19
4.0 CONFIGURATION CONTROLS SYSTEM	4-1
4.1 DRAWING CONTROL	4-1
4.2 PROCEDURE CONTROL	4-1
4.3 UNREVIEWED SAFETY QUESTIONS	4-2
4.4 QUALITY ASSURANCE	4-2

5.0	REFERENCES	5-1
5.1	DOCUMENTS	5-1
5.2	U.S. FEDERAL CODES AND DEPARTMENT OF ENERGY ORDERS/GUIDELINES	5-2
5.3	WESTINGHOUSE HANFORD COMPANY CONTROLLED MANUALS	5-3
5.4	PROJECT HANFORD PROCEDURES	5-4
5.5	WASTE MANAGEMENT FEDERAL SERVICES OF HANFORD PROCEDURES	5-4

APPENDICES

A	340 FACILITY HYDROGEN GENERATION ESTIMATES	A-1
B	CALCULATION OF SPRAY RELEASE	B-1
C	GENII DOSE CONSEQUENCE CALCULATION FOR THE 340 FACILITY	C-1
D	HAZARDOUS MATERIAL RELEASE ESTIMATES	D-1
E	APPROVAL OF INTERIM SAFETY BASIS (ISB) FOR THE 340 WASTE HANDLING FACILITY	E-1

LIST OF FIGURES

1-1.	340 Waste Handling Complex Site Plan	1-6
1-2.	Aerial View of 340 Complex	1-7
3-1.	Onsite and Offsite Radiological Risk Comparison to Guidelines.	3-21

LIST OF TABLES

2-1.	340 Facility Radionuclide Inventory	2-7
2-2.	340 Facility Hazardous Material Inventory	2-8
2-3.	340 Vault Tank Bounding Source Term	2-9
2-4.	340 Facility Solid Waste Bounding Source Term	2-9
3-1.	340 Vault Tank Hazard Category Calculations	3-20
3-2.	Hazard Categorization of 340 B Building Solid Radioactive Waste	3-20
3-3.	Calculation of Plutonium Equivalent Curie/Liter	3-20
3-4.	340 Waste Handling Facility Preliminary Hazard Analysis	3-21
3-5.	Probability Category Definition and Qualitative Accident Severity Levels	3-27
3-6.	Operating Hazards.	3-28
3-7.	Maximum Individual Dose Due to an Aerosol Release of 1 Curie of Each Isotope From the 340 Facility	3-30
3-8.	Dose Consequences for a Spray-Release Accident	3-31
3-9.	Dose Consequences When Burning the 340-B West Building Solid Wastes	3-31
3-10.	340 Facility Safety Enhancement Systems	3-32

LIST OF TERMS

ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
CSER	criticality safety evaluation report
DAC	Derived Airborne Concentration
DOE	U.S. Department of Energy
EDE	effective dose equivalent
ERPG	Emergency Response Planning Guideline
FGE	fissile gram equivalent
FSAR	final safety analysis report
HSRCM	Hanford Site Radiological Control Manual
HEPA	high-efficiency particulate air (filter)
HPT	health physics technician
IOSR	interim operational safety requirements
ISB	interim safety basis
MSC	major subcontractor
PHA	preliminary hazards analysis
Pacific Northwest	Pacific Northwest National Laboratory
PHMC	Project Hanford Management Contract
QA	quality assurance
RL	Richland Operations Office (DOE)
RLW	radioactive liquid waste
RLWS	radioactive liquid waste system
RPS	retention process sewer
RWP	radiation work permit
SSC	structures, systems, and components
TEDF	Treated Effluent Disposal Facility
TEEL	temporary emergency exposure limits
TQ	threshold quantities
TQ _m	threshold quantities modified
TRU	transuranic
USQ	unreviewed safety question
WHC	Westinghouse Hanford Company
WMH	Waste Management Federal Services of Hanford

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1.0 INTRODUCTION AND SUMMARY

1.1 PURPOSE

This document establishes the interim safety basis (ISB) for the 340 Waste Handling Facility (340 Facility). An ISB is a documented safety basis that provides a justification for the continued operation of the facility. The ISB is not intended to be compliant with U.S. Department of Energy (DOE) Order 5480.23, *Nuclear Safety Analysis Reports*, but is written to demonstrate adequate level of safety basis for the 340 Facility operations. The ISB for the 340 Facility documents the current design and operation of the facility. Due to the limited remaining active life for the core of the 340 complex (see sections 1.3 and 1.4), this ISB (now Rev. 2) will continue to serve in place of an upgraded FSAR, per provisions in HNF-PRO-430, *Safety Analysis Program* and HNF-PRO-701, *Safety Analysis Process-Existing Facility*.

The 340 Facility will comply with all applicable state and federal statutes, DOE Orders, rules, and regulations which establish safety constraints on the 340 Facility. The 340 Facility will comply with these regulations to ensure that all activities conducted at the facility proceed in a safe manner.

1.2 BACKGROUND

The 340 Facility (Figures 1-1 and 1-2) is located in the 300 Area of the Hanford Site and historically provided the capability for receipt, temporary storage, and shipment of radioactive and mixed liquid wastes via the radioactive liquid waste system (RLWS). The RLWS is an underground piping system through which radioactive and mixed liquid wastes were gravity drained from Pacific Northwest National Laboratory (Pacific Northwest) and B&W Hanford Company (BWHC) buildings into the 340 Building Vault Tanks. Pacific Northwest generates a majority of the radioactive and mixed liquid wastes in the 300 Area; however, infrequent transfers from B&W Hanford Company (BWHC) and other Project Hanford Management Contract (PHMC) managed facilities also occur. As a result of operating and maintaining this system, some radioactive/mixed solid waste is generated. The solid waste is packaged and temporarily stored before shipment. The 340 Facility also accepts radioactive solid waste from other generators. The 340 Facility does not routinely handle solid transuranic (TRU) wastes and does not receive or store TRU waste from other generators.

The 340 Complex also manages the potentially radioactive, nonhazardous retention process sewer (RPS) liquid waste stream. The 307 Basins (Figure 1-1) serve as a holding/monitoring point for the RPS liquid which discharges to the Treated Effluent Disposal Facility (TEDF) located north of the 300 Area. In the event that the RPS does become radioactively contaminated, monitors in the upstream facilities signal the basin control system to switch RPS inlet flow to a segregated basin. In some cases, the monitor signal activates a valve which retains flow within that facility. The RPS basins are periodically sampled before discharge and are, as appropriate, pumped to the TEDF or loaded into a truck for shipment elsewhere.

The 307 Basins were evaluated apart from this ISB as part of the RPS (Berneski 1998) which determined that this system falls below the threshold for consideration under the requirements of DOE Order 5480.23 (less than a Hazard Category 3 facility as defined in DOE-STD-1027, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23*,

Nuclear Safety Analysis Reports). The evaluation for this process meets the requirements of DOE Order 5481.1B, *Safety Analysis and Review System* and has been issued as a supporting document (Berneski 1995).

1.3 FACILITY STATUS

The 340 Facility still provides temporary storage for radioactive and mixed liquid waste and serves as a staging area for low-level radioactive and mixed solid waste. Actions to deactivate those portions of the 340 Facility associated with radioactive and mixed liquid waste storage (Stordeur and Beckstrom 1998) are partially complete. The 340 Facility is not a routine generator of solid TRU waste and does not require a facility-specific TRU waste certification plan. The facility does generate low-level waste and uses the waste profile system (per WHC 1993). The 340 Facility no longer provides decontamination services to other facilities. Flammable liquids are not allowed in the RLWS, which eliminates the potential for a fire in the 340-A Building and the Vault Tanks. Rags containing solvents, oil, combustibles or flammables are collected in a drum located outside of the 340 Building.

1.4 FUTURE ACTIVITIES

Future projects that modify the 340 Facility will have a preliminary "safety analysis" completed before start-of-construction to document changes to the Facility Hazard Category and dose consequences from postulated accidents. This will ensure that the proposed modifications comply with current safety requirements and safety-related design criteria. As of October 1, 1998, operation of the RLWS was terminated with respect to new waste receipts. Therefore, it is anticipated that no upgrade projects will be pursued, other than minor actions deemed necessary or prudent to place the facility systems and/or areas in a safe and stable condition such that cost effective surveillance and maintenance programs will prevent any unacceptable risk to the public or the environment until the facility can be ultimately dispositioned (Stordeur and Beckstrom 1998).

1.5 CONFIGURATION MANAGEMENT

The PHMC configuration management program (HNF-PRO-1819, *PHMC Engineering Requirements*) addresses design changes and controls with guidelines that cover changes to the facility documents (WHC-IP-1026, *Engineering Practices Guidelines*, WHC 1994a and HNF-PRO-440, *Engineering Document Change Control Requirements*). The facility's cognizant engineers are responsible for identifying any additional documents, drawings, instructions, or procedures that are affected by a facility change. The documents requiring changes may include operating procedures, administrative procedures, and maintenance procedures. These changes require review and approval by Quality Assurance, Criticality safety, Environmental, and independent safety organizations as appropriate.

ISB-003 provides information relating to control of hazards at the 340 Facility (including design, engineering analysis, and administrative) from which the conclusion is made that current and remaining 340 Facility activities can be conducted safely. Changes to these descriptions will be updated at least annually via the engineering change procedure specified in HNF-PRO-440, *Engineering Document Change Control Requirements*. Also, HNF-PRO-701, *Safety Analysis Process-Existing Facility*, requires an annual update for safety analysis documentation.

1.6 SUMMARY OF SAFETY ANALYSIS AND SUPPORTING INFORMATION

The nature of the operations performed and the materials handled in this facility present potentials for nuclear accidents (e.g., exposure to direct radiation and radioactive contamination spread), as well as accidents of a more conventional nature (e.g., industrial accidents). Operation of the 340 Facility is managed as part of the PHMC Waste Management Federal Services of Hanford (WMH).

The hazard analysis presented in ISB-003 establishes the 340 Facility Hazard Category as follows:

- RLWS Storage - Hazard Category 3, 340 Facility Vault Tanks and 340-A Building Tanks.
- Solid Waste Storage - Hazard Category 3, 340 Facility Decontamination Area and the 340-B Building.

The maximum radionuclide inventory found in the dose-consequence calculations and the guidelines of DOE-STD-1027-92 were used to determine the hazard category.

The 340 Facility inventory consists primarily of liquid and solid waste containing mixed fission products and other radionuclides. To remain classified as an isolated facility, per HNF-PRO-334, *Criticality Safety General Requirements*, the 340 Facility must maintain <1/3 of a minimum critical mass of fissionable material. Administrative controls for an accurate, auditable inventory have been established per HNF-PRO-334.

The 340 Facility does not contain Safety Class or Safety Significant structures, systems, and components (SSC), per HNF-PRO-700, *Safety Analysis and Technical Safety Requirements*.

The hazards associated with the 340 Facility have been analyzed in a preliminary hazards analysis (PHA) format in Section 3.0. Annual frequency and severity categories are presented with each hazard. The principle types of hazards considered are industrial, high radiation, radiological contamination, and natural disaster. The bounding hazards were further evaluated for consequence and risk acceptance.

The worst-case hazards that presented vulnerabilities in design or operation were identified as the spray leak in the 340-B East and the solid waste fire in the 340-B West Building. As a result of the changing concentration and distribution of isotopes in the 340 Facility RLWS¹, sufficient inventory could allow an unmitigated spray release to exceed onsite/offsite dose limits or Hazard Category 2 thresholds. The RLWS inventory was simplified into curies of beta/gamma and alpha emitters. A worst-case beta/gamma inventory of 100% ¹⁵⁴Eu is assumed in a conservative approach because it has the highest onsite rem/Ci value. Similarly, ²³⁹Pu has one of the highest onsite rem/Ci values and is assumed to be a worst-case

¹As a consequence of the closure of the RLWS to new waste receipts noted in Section 1.4, the residual inventory is essentially fixed (subject to minor additions from deactivation cleanout tasks). This makes the source term used in the hazard evaluations even more conservative.

alpha inventory. The tables in Section 3.0 assume a worst-case beta/gamma and alpha inventory of ^{90}Sr (^{90}Sr is more conservative for hazard category calculations than ^{154}Eu) and ^{239}Pu , respectively, for hazard category calculations. The calculated releases for the spray-leak, worst-case accident are within the risk-acceptance guidelines. The dose consequences are below the Safety Class and Safety Significant criteria. Operational restrictions of 66 Ci total alpha and 39 kCi total beta/gamma have been established for the vault tank to be pumped into waste tankers, based on the accident scenario and the inventory used in the dose calculations.

Similarly, an unmitigated solid waste fire in the 340-B West Building could exceed onsite/offsite dose limits or Hazard Category 2 thresholds with sufficient inventory. A total curie inventory of 10 Ci beta/gamma and 1.5 Ci alpha were used in the dose calculations in Section 3.0. Once again, ^{154}Eu and ^{239}Pu were used as worst-case beta/gamma and alpha inventories for the dose calculations, and ^{90}Sr and ^{239}Pu were used as the worst-case inventories for the hazard category calculations. The calculated releases for this worst-case accident are within the risk-acceptance guidelines. The dose consequences are below the Safety Class and Safety Significant criteria. Operational restrictions of 1.5 Ci total alpha and 10 Ci beta/gamma have been established for the solid waste based on the inventory used in the dose calculations.

1.7 CONCLUSIONS

The analysis in this document establishes that the 340 Facility is adequately designed and controlled to operate without undue risk to the health and safety of the onsite worker, the public, and the environment. A PHA (Section 3.0) identifies bounding hazards for further evaluation. The accident dose consequences show that possible releases, as a result of a RLWS spray leak in the 340-B East Building and a solid waste fire in the 340-B West Building, are below risk-acceptance guidelines based on the maximum inventories established herein. Operational restrictions in Section 3.0 were established to ensure safe operation. These controls identify an operating envelope in which the 340 Facility's interim safety is based.

In accordance with DOE-STD-1027-92, the 340 Facility RLWS, and Solid Waste Storage have been determined to be Hazard Category 3. Therefore, by providing a safety analysis based on the bounding source terms and the operations and design of the facility, it can be concluded that ISB-003 adequately defines the safe operating envelope. The safety analyses summarized in this document define the authorization basis for performing unreviewed safety question (USQ) screenings and evaluations in accordance with HNF-PRO-062, *Identifying and Resolving Unreviewed Safety Questions* and DOE Order 5480.21, *Unreviewed Safety Questions*.

The conduct of operations for the 340 Facility, with respect to the institutional and safety controls, have been evaluated. This evaluation compared facility-specific procedures that implement the institutional and safety controls established by DOE Orders. Since ISB-003 Rev. 0 was issued, the 340 procedures and administrative control documents have been upgraded to comply with the revisions. It can be concluded that the existing facility procedures adequately implement the PHMC policy and have adequately addressed the 340 Facility conduct of operations.

Figure 1-1. 340 Waste Handling Complex Site Plan.

Figure 1-2. Aerial View of 340 Complex

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2.0 FACILITY DESCRIPTION

2.1 GENERAL

The 340 Facility has supported upstream facilities through the receipt, storage, and shipment of radioactive liquid wastes generated during laboratory operations. The 340 Facility also occasionally received containerized liquid waste from PHMC managed facilities in the 300 Area. Solid waste stored in the 340-B Building Storage Area primarily comes from 340 Facility operations and maintenance activities. Other PHMC 300 Area solid waste generators occasionally use the 340 Facility as a staging area for solid radioactive waste transfers.

The 340 Facility¹ consists of the following five buildings:

- 340 — control room, decontamination and repackaging area, mechanical equipment room, change rooms, truck lock area, and an operator's office
- 340-A — above ground auxiliary RLWS tank storage
- 340-B East — [former] railroad car loading station
- 340-B West — radioactive solid waste storage area (340-B East and 340-B West are one building, divided by a wall)
- MO-036 — office trailer
- MO-741 — health physics technician (HPT) office and survey station.

The 300 Area RLWS, consisting of encased underground piping and ancillary equipment, connected the 340 Facility to upstream facilities. The RLWS historically accepted radioactive liquids and mixed waste. These liquids were gravity drained into the tanks within the 340 Building Vault. Transfers, if any, to the 340-A tanks required pumping via the vault tanks. All 340 tanks (vault and above ground) and interconnecting piping have undergone and satisfactorily completed containment integrity checks (WHC 1996a).

2.2 PROCESS/OPERATION

The 340 Facility manages the RLWS. Radioactive waste was received via underground piping that connects² the 324, 325, 326, 327, and 329 Buildings to the 340 Vault Tanks, or was transported to the facility in containers for addition to the 340 Vault Tanks. Containerized waste receipts outside the RLWS system entered the vault tanks either via the sump in the 340 decontamination area (see 2.3.1), or the lines in the 340-B load-out facility (2.3.3). Examples of sources for containerized deliveries were lysimeter wastes and the 309 facility decon solutions. "Containers" were anything from small carboys and 55-gallon drums to 5000 gallon tanker trucks.

¹As differentiated from the 340 "Complex", which includes the 307 basins, 3707-F, and the RPS connecting system, outside the scope of this ISB as indicated in section 1.2.

²While existing in-line valves were closed as of October 1, 1998, the lines will be cut and capped.

The RLWS equipment is constructed of stainless steel pipe and tanks. Buried steel pipe is encased in fiberglass-reinforced plastic pipe. Each section of the RLWS passes through a concrete valve box, which allows for the isolation of the flow. These valve boxes are equipped with a sump, and sump leak detector. The deactivation of the RLWS transfer piping at the generator building negate the need for maintenance of the valve boxes and associated instrumentation i.e. sump leak detectors.

The facility accumulated and stored waste in two 57 kL (15,000 gal) below ground tanks. One tank was always online to receive waste, which allows the other tank to be isolated for transfer. The tanks, known as TK-1 and TK-2, are equipped with mechanical paddle agitators for mixing and centrifugal pumps for waste transfer. The liquid level in the tanks is recorded using a bubbler-type, liquid-level indicator. Each tank has two liquid-level alarm points: influent and high level. The tanks are equipped with an automatic function that may be used to automatically transfer the contents of either tank to the 340A auxiliary storage should the high level be exceeded. In addition, TK-1 and TK-2 are connected by an overflow line. These tanks may be used during interim/final 340 facility deactivation to consolidate decontamination liquids prior to disposal.

Before transferring waste out of the 340 Facility, Operations scheduled a rail tanker for delivery to the 340-B East Building. The 340-B East Building is a pre-engineered metal building with a high roll-up door and a rail spur leading into the east side of the building. The building is designed to handle the 76 kL rail tankers used on the Hanford Site. To prepare waste for shipment, the transfer tank is isolated from incoming waste and sampled. The tank contents are agitated to homogenize the waste before samples are taken. The samples are sent to a laboratory for analysis. The sample results dictated how much waste could be safely loaded into a rail tanker and shipped to the 200 Area.

A rail tanker was then received into the east side of the 340-B Building, which has its own ventilation system. The tanker was dose rated and surveyed for radioactive contamination and, if acceptable, the tanker was connected to the RLWS. Although valves in the 340-A and 340-B Buildings are manually operated, much of the loadout operation — position vault valves, pump actions, and liquid level indication alarms — takes place remotely in the 340 Building control room. The rail tanker was loaded and shipped to the 204-AR unloading facility located in the 200 East Area for ultimate transfer to the double-shell tanks³.

Internal waste transfers are done from the 340 Building control panel. Many of the transfers are automatic functions eliminating the need to establish valve line ups. Waste can be transferred between TK-1 and TK-2, from TK-1 or TK-2 to above ground auxiliary storage, and from above ground storage to TK-1 or TK-2. The transfer piping from the above ground tanks to the vault tanks is currently valved out as part of the October 1, 1998 deactivation and isolation of the vault tanks. However, if clean out activities require the use of the transfer pipes the valves can be opened.

³ With the deactivation of the rail tankers, liquid waste transfer from the 340 facility will be by truck tankers. However, loadout activities will be consistent with the above description and final disposal options other than 204-AR may be viable.

Radioactive and mixed solid wastes are generated in the 340 Facility during the performance of routine operations, maintenance, and decontamination activities. Waste generated at the facility are typically low-dose rate waste, and are classified as low-level waste. The majority of the waste is packaged in drums or boxes and transported by truck to the 200 Area for burial or storage.

2.3 BUILDINGS

As deactivation of the 340 facility progresses various systems described below may become obsolete and unnecessary for routine operations. The deactivation activities will be controlled by administrative procedure HNF-IP-1000, section 2.2 "340 Facility Equipment/System Deactivation Lay-Up Status and Control", derived from HNF-PRO-072. The lay-up procedure allows for system reactivation if warranted and relevant precautions are exercised.

2.3.1 340 Building

The 340 Building is an above ground structure housing the 340 Facility operator office, change rooms, control room, equipment decontamination room (with a sample hood), mechanical equipment room, and truck lock. Below ground, the 340 Vault contains two RLWS 57 kL (15,000 gal) tanks and a valve pit containing valves and piping partitioned off from the main vault. The primary above grade structure is steel and sheet metal attached to a steel frame mounted on a concrete foundation.

The 340 Building is provided with a ventilation supply system (including evaporative cooling and electric heating), and an exhaust system that serves all of the rooms except the control and mechanical equipment rooms. The exhaust system consists of two-stage high-efficiency particulate air (HEPA) filters, exhaust fan, and a stack that is located above the rest rooms.

The decontamination area contains a radioactive liquid waste (RLW) sample hood and a large open work area in the room for decontamination. A steel grating over a shallow concrete trough services the open work area. The grating could easily be replaced if it could not be decontaminated. The trough drains into a sump in the corner of the bay. In the past, wash solutions used for decontamination and water rinse were pumped from the sump directly to a vault tank.

The control room and a cabinet located next to the stack contain effluent monitoring and control instrumentation for the RLWS loadout system, valve controls, the main electrical control panel, and dryer for the compressed air system. The control room has heating and cooling capabilities and is accessed only through an outside door.

The 340 Building control room annunciator panel has been replaced with computerized monitoring and alarming capabilities. Remote alarm capability was expanded by routing local alarm indication through the 300 LEF computer network. Some alarming conditions (radiation alarms) are annunciated at the source, through the alarm horn in the 340 control room and building 310 TEDF.

The small mechanical equipment room south of the control room, accessible only via an outside door, contains instrument air compressors (for the bubbler system, air operated valves), roof mounted evaporative cooler, and vacuum pumps for the stack monitors.

The 340 Building Vault walls are constructed of concrete, with concrete shielded cover blocks and access holes. A hatch is provided in the roof allowing entry for inspection and maintenance. The valve pit, accessible through a separate hatch, is located next to the vault and contains valves, piping and a sample/backup pump. Hatches to the vault and valve pit have been covered with locked penthouses. The 340 Building Vault and vault tanks are serviced by a common exhaust system, which discharges through a stage of dust stop filters, two-stage HEPA filters, and two exhaust fans located just east of the vault. The capability exists for the addition of activated charcoal adsorbent into the exhaust treatment train. The charcoal filters were used in the past to help control potential Iodine-131 releases associated with activities in the 327 Postirradiation Testing Laboratory.

2.3.2 340-A Building

The 340-A Building is constructed of steel panels attached to a steel frame mounted on a concrete foundation. The floor is set below grade level. The floor and lower portions of the walls are made of concrete. The floor contains a dry sump, which used to drain back to the 340 Building Vault Tanks. The sump is currently capped as part of the vault tank isolation. Six 30 kL (8,000 gal) stainless steel tanks are housed in the building to provide reserve capacity for RLW. There is no building ventilation and the tanks are ventilated through the vault ventilation system.

2.3.3 340-B Building

The east side of the 340-B Building houses a railroad car receiving and loadout station. The west side of the building has a 90-day solid mixed waste pad, and an area for radioactive solid waste, contained in steel drums and/or plywood or metal boxes. Other nonwaste items are also stored in the west side of 340-B Building. This building is constructed of steel panels mounted on a concrete foundation and is divided down the center by a concrete shielding wall extending out and around the south end of the building toward the east. The shielding wall is 12 feet high, so the distance between the top of the wall and the rafters is covered with sheet rock.

The liquid waste loading station is in the east side of the building. A sump is located between the railroad tracks and drains to the 340 Building Vault Tanks. The sump is currently valved out as part of the isolation of the vault tanks, however the valves may be opened to facilitate the 340-B east cleanout. Flexible vent and fill lines are provided that were designed to couple with the railroad car. A different waste tanker will require new adapters for the vent and fill lines. Water flushing equipment is provided to clean the exterior of the railroad cars. The sump drain line valve is interlocked with the liquid waste transfer pumps to prevent inadvertent pressurization of the drain line and back filling of the sump.

The east side of the 340-B Building is serviced by a ventilation system (located in the west side of the building) consisting of a HEPA filter, vent fan, and stack. This ventilation equipment is primarily operated when a railroad car is in the loading station and connections are being made or broken, or the loading operation is in progress. The west side of the building does not have a supply or exhaust ventilation system. Electric ceiling heaters are used for heating when required.

2.3.4 MO-036 Building

The MO-036 building is a double-wide trailer inside the 340 Facility fencing west of the 340-B building. This trailer provides offices for facility personnel.

2.3.5 MO-741 Building

The MO-741 building is a small single-room trailer south of MO-036. This trailer provides an office for Radiological Control personnel, and contains two personnel contamination monitors (PCM-1B).

2.4 RADIONUCLIDE AND HAZARDOUS MATERIAL INVENTORIES

The 340 Facility does not routinely handle or store hazardous chemicals within the buildings. However, hazardous material is present in the radioactive liquid waste. A data base developed over a 5-year period — 1989 through 1993 — for the 340 Facility recorded radionuclide and hazardous material concentrations for tank waste liquids before the waste was shipped from the 340 Facility via rail tank car to the 204-AR Unloading Facility in the 200 East Area. The maximum value for each radioactive constituent was entered into a spreadsheet and used for the radionuclide source term. These maximum concentrations are shown in Table 2-1. Shipments made since the 5-year waste summary was first prepared have not exceeded Table 2-1 in terms of the maximum values depicted. These values are still considered valid for Rev 2; and therefore continue to serve the purpose of demonstrating the conservatism of the bounding source term. The hazardous material source term is presented in Table 2-2.

2.4.1 340 Vault

The bounding source term for the 340 Building Vault Tanks assumes the sending tank contains its maximum 57 kL volume of a worst-case inventory. Because the RLW inventory is dependent on the work in upstream laboratories, radionuclide concentrations may change. The worst-case inventory has been established based on the maximum inventory recorded in the past (Table 2-1). These historical values had been increased to account for possible excursions when the RLWS was active, and these totals are shown in Table 2-3. Therefore, Table 2-3 defines the bounding source term for liquid waste collected in a vault tank for transfer into a rail car. This bounding source term has been simplified to define total beta/gamma curies and total alpha curies in a full vault tank. The isotopic break-out depicted in Table 2-1 serves as the basis for substituting appropriate and realistic, worst-case nuclides for alpha or beta/gamma totals into the particular release scenarios and dose models used in various section 3.0 evaluations.

2.4.2 340 Building

The 340 Building inventory consists of radioactive and mixed solid waste and radioactive liquid waste in the decontamination area. The decontamination area in the 340 Building is a satellite accumulation area for solid waste, which is moved to the storage pad in the 340-B West Building. The sample sink and hood allow for processing samples from the vault tanks. Drains from the sample sink and hood return liquid to the sump. Decontamination solution drains through floor grating to trenches and then to the sump. The sump has a

pump that could transfer liquid waste back to the vault tanks, a path now deactivated as part of the RLWS isolation.

2.4.3 340-A Building

The six cross-connected tanks in the 340-A Building are not currently used for waste storage. After a recent (3/98) cleanout, a small heel of sludge is assumed to exist in the portion of the tank below the piping connection to the tanks. An accumulation of fissile material is conservatively estimated to be < 11 grams in the sludge. This range reflects: 1) the results of grab samples of sludge, 2) an estimate based upon process knowledge of the insolubility of plutonium, 3) the amount of plutonium that passed through the tanks when they were in continuous use, and 4) the post cleanout residue assumption. The tanks are valved out and the remaining fissile material will not readily migrate from the facility.

2.4.4 340-B Building

The west side of the 340-B Building contains solid radioactive and mixed waste storage areas separated from the loadout facility by a 30 cm (12 in.) thick concrete shield wall. Waste from the 340 Facility is collected and packaged where it occurs, then moved to the 340-B West Building for storage and shipment. Shipping records for solid waste generated in the 340 Facility during the years 1989 through 1993 show that the maximum 208-L (55-gal) drum contained 3.08×10^{-3} Ci and the maximum plywood box contained 7.2×10^{-4} Ci.

Higher curie levels were selected for the solid waste storage areas to define the bounding inventory. These higher curie values will allow the facility to operate without exceeding the approved operating envelope. Solid waste storage area values selected and used in the accident analysis are 10 Ci beta/gamma and 1.5 Ci alpha as shown in Table 2-4. The accident analysis in Section 3.0 used these estimates of worst-case beta/gamma and alpha inventories for the onsite and offsite dose calculations.

The east side of 340-B is used primarily for RLW transfers and will contain RLW in the tanker during transfer operations. Occasionally a high-radiation, solid waste drum containing contaminated equipment (i.e., valve bodies, valve diaphragms, pump components, flanges, or sections of pipe) will be packaged in a shielded drum. Any such waste that would be considered a worker radiation exposure hazard is stored in the 340-B East Building to reduce exposure to workers.

Table 2-1. 340 Facility Radionuclide Inventory.

Item	Maximum Ci/L*
Total alpha	7.40 E-04
Total beta	2.77 E-02
⁵⁴ Mn	7.66 E-07
⁶⁰ Co	1.85 E-06
⁹⁰ Sr	1.07 E-03
⁹⁵ Zr	2.25 E-06
⁹⁵ Nb	4.01 E-06
⁹⁹ Tc	4.73 E-08
¹⁰⁶ Ru	4.96 E-05
¹²⁵ Sb	3.96 E-05
¹³⁴ Cs	6.87 E-05
¹³⁷ Cs	2.37 E-02
¹⁴⁴ Ce	3.15 E-04
¹⁵² Eu	4.49 E-03
¹⁵⁴ Eu	7.90 E-03
¹⁵⁵ Eu	1.90 E-03
¹⁸² Ta	1.14 E-05
²³⁴ U	1.50 E-07
²³⁵ U	1.88 E-09
²³⁶ U	1.23 E-08
²³⁸ U	5.98 E-08
²³⁸ Pu	2.72 E-04
^{239/240} Pu	5.81 E-05
²⁴⁰ Pu	7.85 E-05
²⁴¹ Pu	7.48 E-03
²⁴² Pu	2.35 E-07
²⁴¹ Am	6.26 E-04
^{243/244} Cm	1.67 E-06

*Based on rail car shipping concentrations, 1989 through 1993.

Table 2-2. 340 Facility Hazardous
Material Inventory.

Item	Maximum mg/L*
Al ⁺³	3.20 E+03
As ⁺³	4.15 E+00
Ba ⁺²	1.14 E+01
HCO ₃ ⁻	5.38 E+02
Cd ⁺²	6.64 E+00
CO ₂	5.50 E+02
CO ₃ ⁼	2.91 E+01
Cl ⁻	3.40 E+03
Cr ⁺²	6.85 E+01
F ⁻	1.55 E+01
OH ⁻	4.25 E+02
Pb ⁺²	8.72 E+01
Hg ⁺²	1.25 E-01
NO ₃ ⁻	3.42 E+04
NO ₂ ⁻	3.01 E+03
PO ₄ ⁻³	7.27 E+03
Se ⁼	6.23 E+00
Ag ⁺¹	4.46 E+00
Na ⁺	2.07 E+03
SO ₄ ⁼	1.61 E+02
TOC	3.63 E+03

*Based on rail car shipping concentrations,
1989 through 1993.

TOC = total organic carbon.

Table 2-3. 340 Vault Tank Bounding Source Term.

Isotope	Facility Inventory in Ci/L ⁽¹⁾	Ci/Tank 57 kL (15,000 gal)
Beta/Gamma	6.90 E-01	3.93 E+04
Alpha	1.16 E-03	6.61 E+01

⁽¹⁾ As noted in Section 2.4.1, the historical maximum concentrations presented in Table 2-1 were further increased here (conservatively accounting for possible future excursions over past observances) to function as the bounding source term. Total alpha and beta/gamma values are presented because these gross parameters conform to the typical focus of both monitoring and laboratory screening analyses. In the next column these concentrations are in turn multiplied by the maximum vault tank capacity, 15,000 gallons.

Table 2-4. 340 Facility Solid Waste Bounding Source Term.⁽¹⁾

Isotope	Worst-case Inventory of Solid Waste Ci
Beta/Gamma	10.00 E+00
Alpha	1.50 E+00

⁽¹⁾ As noted in Section 2.4.4, "Shipping records for solid waste generated in the 340 Facility during the years 1989 through 1993 show that the maximum 208-L (55-gal) drum contained 3.08×10^{-3} Ci and the maximum plywood box contained 7.2×10^{-4} Ci. Higher curie levels were selected for the solid waste storage areas to define the bounding inventory. These higher curie values will allow the facility to operate without exceeding the approved operating envelope." Similar rationale to that given in the above Table 2-3 footnote explains use of gross, or total, radionuclide levels.

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3.0 INTERIM SAFETY BASIS EVALUATION

In order to limit risk from hazards and thereby conclude that the 340 Facility can be operated safely, the following sections reference existing safety analysis documents and provide information that identifies and evaluates hazards, identifies controls and/or mitigation systems, and identifies the necessary administrative controls and/or operational restrictions.

3.1 HAZARD ANALYSIS

This section summarizes the hazards of operating the 340 Facility using a PHA of the facility and its appurtenances. This section also evaluates the source terms — radiological and nonradiological — for the areas of the facility containing hazardous materials, determines the hazard categories for the areas within the facility, identifies the potential accident sequences based on the worst-case hazards from the PHA, estimates the potential consequences of the accidents, and provides a quantitative evaluation of the risk for the accidents.

3.1.1 Hazard Categorization

This section identifies the hazard category for each part of the 340 Facility using the bounding source term for that part of the facility. The hazards analysis categories for the 340 Facility are based on the "graded approach" and "facility segmentation" concepts of DOE-STD-1027-92. The "graded approach" ensures that the thoroughness of the documentation be graded or proportioned commensurate with the magnitude of the hazards, the complexity of the facility, and the stage or cycle of the facility. The "facility segmentation" concept in accordance with DOE-STD-1027-92 applies to inventories of hazardous materials where facility features preclude collocating the material and material interaction during a postulated accident (340 Complex Fire Hazard Analysis [Schermerhorn 1996] also supports facility segmentation). There is a physical separation between buildings in the 340 Facility. Also, the 340-B Building has a concrete wall that separates the building into a 340-B East area and a 340-B West area. These physical separations and concrete wall provide barriers between the hazardous sources that essentially prevent the interaction of hazardous materials during a postulated accident. Also, an automatic sprinkler system has been added to the east and west side of 340-B since the original ISB evaluation. The hazard categorization is based on radiological hazards in accordance with DOE-STD-1027-92.

3.1.1.1 Radioactive Liquid Waste System Hazard Category. The RLWS hazard category is established in the following subsections.

340 Vault Tanks — A spray release, while filling the railroad tank car from one of the 340 Vault Tanks, is the credible accident considered for hazard category determination. Since the 20,000 gallon capacity rail tanker could accommodate an entire vault tank, the consequences of the spray release scenario effectively bounds any similar release that might occur during loadout to a similar size or smaller truck tanker. The Ci/threshold quantities modified (TQ_m)-Ci for the vault tanks, based on the modified Hazard

Category 2 threshold quantities of one vault tank (57 kL) is shown in Table 3-1 (note discussion in Section 3.1.2). The total $\text{Ci/TQ}_{\text{m}}\text{-Ci}$ of 2.66×10^{-02} is less than 1, placing the vault tanks within Hazard Category 3.

340-A Tanks — The 11 fissile grams in the 340-A Tanks, using the specific activity for ^{239}Pu of 6.2×10^{-2} Ci/g becomes 0.68 Ci ^{239}Pu , with a Hazard Category 2 threshold quantity (TQ) of 5.6×10^1 Ci giving a $0.68 \text{ Ci} / 5.6 \times 10^1 = 1.2 \times 10^{-2}$ Ci/TQ-Ci, which is less than 1, placing the 340-A Tanks in Hazard Category 3. A plutonium inventory in the 340-A tanks equivalent to the facility limit of 1/3 of a minimum critical mass will not exceed the calculations for a Hazard Category 3 facility.

3.1.1.2 Solid Waste Area Hazards Category. A solid waste fire that would consume the entire inventory (Table 3-2) results in a total Ci/TQ-Ci of 2.72×10^{-02} , based on Hazard Category 2 threshold quantities per DOE-STD-1027-92, placing the solid waste in Hazard Category 3.

3.1.2 Hazard Categorization Methodology

If a facility contains greater than Hazard Category 2 TQ, as defined in DOE-STD-1027-92 and shows the potential for significant onsite consequences, the facility is defined as Hazard Category 2. Attachment 1 of DOE-STD-1027-92 notes that if the ratio of the quantity of each isotope to its Hazard Category 2 TQ is ≤ 1 and the sum of the ratios for each isotope in the facility is ≤ 1 , then the facility is Hazard Category 3. If the sum is greater than 1, the facility is defined as Hazard Category 2. The following equation defines how the sum of the ratios is calculated:

$$\text{Sum of ratios} = \frac{\sum_{i=1}^j \text{Inventory quantity of isotope}}{\text{Category 2 threshold quantity of isotope}}$$

3.1.2.1 Radioactive Liquid Waste System. The hazard categorization calculations for the 340 Vault Tanks (Table 3-1) are based on DOE-STD-1027-92, paragraph 3.1.2, and the spray-release accident described in Section 3.1.4.1 of this document. DOE-STD-1027-92 allows for modification of the threshold limit values by replacing the release fraction assumed in DOE-STD-1027-92 with the release fraction calculated for the worst-case unmitigated accident. The material at risk consists of the contents of one of the vault tanks (57 kL). The calculated respirable volume released in the spray is 5.11×10^{-4} kL (Appendix B). The release fraction is determined by dividing the respirable volume released by the material at risk (5.11×10^{-4} kL / $57 \text{ kL} = 8.96 \times 10^{-6}$). The release fraction (8.96×10^{-6}) was divided by .001 (the release fraction assumed for liquids on page A-9, Item 4 of DOE-STD-1027-92) to give 8.96×10^{-3} . The Hazard Category 2 TQ values were divided by 8.96×10^{-3} to obtain the modified threshold values TQ_{m} . The modified threshold values TQ_{m} were used to calculate the Ci/TQ_{m} for ^{90}Sr and ^{239}Pu in Table 3-1. The sum of ratios of the Ci/TQ_{m} s, 2.66×10^{-2} Ci, is less than 1.0 and places the RLWS in Hazard Category 3. Table 3-3 contains the plutonium equivalent curie (PE-Ci) calculations based on the alpha inventory in Table 2-1.

The 340-A Tanks were considered separately from the 340 Vault tanks because the immobility of the inventory precludes it from consideration in the vault tank accident scenario. However, it still contributes to the hazard category.

3.1.2.2 Solid Waste. The bounding radiological source term for the solid waste is shown in Table 2-4. The isotopes in Table 2-4 were used in an accident scenario that involved a solid waste fire that consumes the entire inventory. Dividing the curie inventory by the Hazard Category 2 TQ of DOE-STD-1027-92 results in a total Ci/TQ-Ci of 2.72×10^{-02} (Table 3-2). The Ci/TQ-Ci is less than 1, placing the solid waste in Hazard Category 3.

3.1.3 Preliminary Hazards Analysis

The 340 Facility PHA shown in Table 3-4 systematically lists the hazard/energy sources for the facility, identifies the potential accident and sequence of events leading up to the accident, and tabulates the target/potential consequences of the accident. Table 3-4 includes both engineered and administrative mitigating features, as applicable, for each potential accident. Table 3-4 also provides an estimate of the qualitative accident severity levels using the annual frequency and severity category descriptions in Table 3-5. From the longer list of accidents in Table 3-4, a small number of worst-case accidents were selected that represent accidents having the highest probability and greatest severity coupled with the highest dose consequences. These accidents are then evaluated for both onsite and offsite dose consequences. By demonstrating that the 340 Facility can be operated safely, assuming worst-case unmitigated releases from the highest severity/highest probability accidents, it can be concluded that the facility can also be operated safely assuming unmitigated accident releases of the other lower probability/lower severity accidents.

The worst-case accidents having the highest severity/highest probability that were selected for further evaluation in this section include:

- A spray release occurs upon failure of the pipe or pipe connection while pumping liquid into a railroad tank car (Table 3-4, II.A.5)
- A fire occurs in the 340-B West dry-waste storage area (Table 3-4, I.A.2.g).

3.1.3.1 Fire. A fire in one of the 340 Facility buildings could result in equipment and structure damage, and personnel injury. There is also a potential for radiological and chemical release based on inventory. Flammable quantities of liquids are not allowed in the RLWS, which eliminates the potential for a fire in the tanks in 340-A and the vault. The 340 Facility contains solid mixed waste in the 340 Building and the 340-B Building. Both the 340 and 340-B Buildings have a fire detection and suppression system. The majority of the solid waste is stored in the 340-B West Building with some waste in the 340-B East Building. The amount of solid waste stored in the 340-B East Building is minimal and nonroutine. The release due to a solid waste fire in the 340 Building or 340-B East Building would be insignificant, compared to a solid waste fire in 340-B West Building.

The 300 Area Steam Plant Replacement project, DOE/EA-1178, *300 Area Steam Plant Replacement, Hanford Site, Richland, Washington* (DOE/EA 1997), provides

packaged natural gas fired steam boilers and an underground natural gas pipeline system designed to improve steam system reliability and reduce energy expenditures. Use of natural gas as a fuel supply and packaged steam boilers introduces the risk of boiler explosion, natural gas detonation, fire exposure, flame jets from pipeline rupture and asphyxiation. The effects of fire and explosion scenarios involving the boiler and natural gas pipelines were quantified to determine minimum separation distances that would reduce the risks to nearby nuclear facilities, PNNL-11660, *Minimum Separation Distances for Natural Gas Pipeline and Boilers in the 300 Area, Hanford Site* (Daling 1997). The risk is acceptable for the 340 Facility buildings because the distance to the nearest boiler annex is approximately 180 m (greater than the minimum separation distance of 25 m) and the nearest pipeline is approximately 60 m (greater than the minimum separation distance of 12.5 m) (DOE/EA 1997 and Daling 1997). Also, the majority of the radioactive inventory for the 340 Facility is contained in the underground vault tanks which would be unaffected by the shock pressure from an exploding boiler (underground concrete walls would be an effective shock barrier).

3.1.3.2 Corrosion. The 340 Facility does not provide long-term storage of solid or liquid waste drums, eliminating the potential for containment failure, as a result of corrosion. There is a potential for corrosion in the RLWS pipelines, the 340 Vault Tanks, and the 340-A Tanks. However, the pipelines are encased and the vault tanks have a concrete vault. Vault, and sump leak detectors are also present. Corrosion presents an insignificant potential for hazard to the onsite/offsite person.

3.1.3.3 Potential Kinetic Energy. There is a potential for solid or liquid waste release as a result of drum/container impacts by heavy equipment, sharp/hard equipment, or by dropping. However, only low levels of waste are allowed and the effect of a spill would be local with a minimal airborne release. A helicopter or plane crash would also produce minimal airborne release and is considered incredible, even if a helicopter rescue service is factored in. The highest potential for an airborne release would be a pipe failure while pumping liquid waste to the tank car. This failure could produce a spray release and affect the onsite/offsite individual.

3.1.3.4 Natural Occurrences. Regarding natural forces criteria, early Hanford Site structures were designed for static, vertical live and dead loads, and have limited seismic resistance because the Site was classified as a low-intensity, low-frequency earthquake zone. Later, a classification was made to Zone 2B on the Uniform Building Code Seismic Risk Map, and currently the Site is considered an area of moderate to low seismicity (*Uniform Building Code*, UBC 1991). Some seismic resistance is inherent in structures because of design for winds (lateral forces), nonstructural elements (reserve strengths), stability requirements, limits on deflections, and safety factors included in design and working stress values for materials. Buried structures have considerable inherent earthquake resistance because they are constructed to support the backfill, and the backfill will restrict relative motions between the structures and ground during an earthquake.

The result of natural occurring events such as floods, earthquakes, wind and volcanic activity present only minor releases and are considered insignificant compared to the spray release of the RLWS inventory.

3.1.3.5 Stack Release. The 340 Vault and vault tanks are serviced by a common exhaust system discharging through a stage of dust stop filters, two stages of HEPA filters, and two exhaust fans. The capability exists for the addition of activated charcoal adsorbent into the exhaust treatment train. Filter change-out occurs on an as needed basis indicating that neither radionuclide buildup nor dust buildup on the filters is significant. A recent radiation measurement for the HEPA filter housing of 80 mrem/hr after more than 6 years of continuous operation further suggests that radiation buildup in the HEPA filters is moderate. Therefore, HEPA filter system degradation or failure is not a significant hazard and is bounded by the spray release accident. The charcoal filters were used in the past to help control potential I-131 releases associated with activities in the 327 Postirradiation Testing Laboratory.

3.1.3.6 Loss of Utilities. Loss of electrical utilities would result in the loss of stack fans, monitoring, transfer pump operation, tank level indication, and alarm capabilities. Although active ventilation and monitoring would be lost, passive ventilation through the HEPA filters would exist. Loss of transfer pump operations would only eliminate the potential for a spray leak. Loss of tank level indication and alarm capabilities would not affect the integrity of secondary containment. Also, waste handling operation would be curtailed. Therefore, there would be no significant safety related hazard to the onsite/offsite personnel.

Loss of electricity could result in a 340-A Building RLWS pipe rupture during freezing weather, if the transfer lines were reactivated and full of liquid. A pipe rupture will cause subsequent transfers of liquid waste to drain to the 340-A sump where the leak would be detected. The 340-A sump will drain to vault tank 2 if the isolation of the vault tanks is terminated and the sump plug is removed. Therefore, a rupture of the transfer lines would not cause a release to the environment.

Loss of process water would result in the loss of pump seal water, which could burn up the pumps when running. The failure of the pumps would not allow liquid transfers to take place. However, the pumps are interlocked to shut down on low seal water pressure. This is not a significant hazard to the onsite/offsite person. The pump seal water is currently valved out to preclude the potential for a source of water into the vault tanks. The seal water can be turned on when a need for the pumps is identified.

3.1.3.7 Radiation. Solid waste in the 340-B West Building is monitored by the Health Physics Technician (HPT). High-level radiation areas are controlled and present an insignificant hazard to the onsite/offsite person.

The 340 Facility maintains less than 1/3 of a minimum critical mass to maintain its status as an isolated facility per HNF-PRO-334 and the 340 CSER (Miller, 1996). The 340 Facility administratively controls criticality concerns in accordance with HNF-PRO-334 and DOE Order 5480.24, *Nuclear Criticality Safety*. Therefore, adequate criticality controls are maintained and additional controls are not necessary.

Radiolysis of liquid in the vault tanks generates hydrogen gas which rises to the liquid surface, is diluted to safe concentrations by the process vent flow, and exits the facility through the ventilation stack. Estimates of process air flow rates and hydrogen generation rates using the tank bounding

source term (Table 2-3) demonstrate that the hydrogen in air concentration of no greater than .01 volume percent is much less than the lower hydrogen flammability limit - 4 volume percent (Letter, *340 Facility Hydrogen Generation Estimates*, 1998, Appendix A). In addition, the ventilation system backup exhaust fan and absence of ignition sources provide additional assurance that hydrogen deflagration has a low potential for occurrence. Also, the current curie content of waste in the vault is much less than the bounding source term (Appendix A). Assuming loss of power to the exhaust fans, the hydrogen in air concentration would reach 50% of the lower explosive limit in about 10 days (Appendix A) using Table 2-3 source term. This gives operators enough time to provide emergency power to the exhaust fans and maintain safe hydrogen concentrations. Therefore, adequate controls assuring safe hydrogen in air concentration are maintained and additional controls are not needed.

3.1.4 Accident Scenarios and Consequences

The types of operating hazards that could cause a significant release from the 340 Facility are limited. Table 3-6 contains a listing of the potential hazards for this facility, including a listing of the mitigating features and administrative controls used to ensure safe operation. Considering the preliminary hazards identified in Table 3-4 and the potential operating hazards in Table 3-6, a spray release in the 340-B East Building while pumping to a tank car and a solid waste fire in 340-B West Building are the worst-case potential accidents identified for further evaluation.

3.1.4.1 Spray Release from 340 Vault Tanks. A spray release while filling the railroad tank car from the 340 Vault Tanks is the most likely significant accident. Because the vault tanks are not under pressure and are underground, a spray release from the piping can only occur while the transfer pumps are operating. One tank must be valved to receive waste and automatic transfers from one tank to another do not occur while filling the tank car. Therefore, when one tank has been emptied, it is necessary to shut down the transfer pump and switch the valving to the other tank, making the maximum duration of a spray leak the time it takes to pump out one full tank. Also, because of administrative sampling requirements, typically only the contents of a single tank are transferred per railcar shipment.

Each of the vault tanks have a 57 kL (15,000 gal) maximum capacity, and the pump has a capacity of 380 L/min (100 gal/min) with a head of 16 m (53 ft). The maximum time a spray release can occur is 150 min (15,000 gal ÷ 100 gal/min = 150 minutes). This time will be used for the spill ignoring the periodic leak inspections.

The pump head of 16 m (23 psi) and friction losses for the pipeline crack were used in a spreadsheet to calculate the total liquid released and the respirable liquid released using the equations in WHC-SD-GN-TI-30003 (WHC 1992), see Appendix B. No credit was taken for loss in head pressure from the pump to the tank car, about 4.5 m (15 ft), the building, or leak inspections. The spreadsheet calculations for a pipeline crack 7.62 cm (3 in.) long and 1.94×10^{-2} cm (7.65×10^{-3} in.) wide gives a 1.07 m^3 (275 gal) spray release, with a respirable liquid release of $5.11 \times 10^{-4} \text{ m}^3$ (0.132 gal). Using the bounding source term, as established in Section 2.4, the X/Q values from the *Hanford Environmental Radiation Dosimetry System*

Software System, GENII (Napier et al. 1988), dose-consequence calculations (Appendix C and Table 3-7), and the respirable release from above, Table 3-8 shows the onsite and offsite dose estimates of 3.92 rem EDE and 3.30×10^{-01} rem EDE, respectively.

The corresponding release of chemicals for the spray-leak accident is shown in Appendix D. The release flow rate of $5.68 \times 10^{-8} \text{ m}^3/\text{s}$ was multiplied by the onsite X/Q , $4.21 \times 10^{-2} \text{ s/m}^3$ and offsite X/Q , $3.77 \times 10^{-3} \text{ s/m}^3$, and the highest hazardous material inventory (NO_3^- , $3.42 \times 10^4 \text{ mg/L}$) to obtain the release in mg/m^3 . The chemicals released in the spray are well below ERPG levels, per HNF-PRO-704.

The loadout of liquid waste, with a composition bounded by the source term (Table 2-3), to any container with an operating volume less than 57 kL (15,000 gal) utilizing the vault pump at the 380 L/min (100 gal/min) capacity can not exceed the dose-consequence calculations. Also, the consequence of the release of chemicals can not exceed the ERPG values.

3.1.4.2 Fire Involving Solid Waste. The west side of the 340-B Building is both a 90-day solid mixed waste pad, and a radioactive waste storage area, separated from the loadout facility by a 30 cm (12") concrete shield wall. Waste from the 340 Facility is collected and packaged where it occurs, then moved to 340-B West Building for shipment. The solid waste stored on the 340 Building 90-day hazardous waste pad and the 340-B West Building 90-day solid mixed waste pad typically consists of cleaning materials, personal protective equipment and clothing, worn out equipment, material from cleanup activities, and tools that cannot be completely decontaminated and released.

Because the waste is solid and stored in boxes or drums the worst-case accident would be a fire, assuming sufficient combustible contents. Possible onsite and offsite releases were estimated, using an airborne-release fraction (ARF) of 5×10^{-4} from *Increment of Analysis: An Estimate Of Airborne Release Of Plutonium From Babcock and Wilcox Plant As A Result Of Severe Wind* (Mishima et al. 1978). The bounding source term established in Section 2.4, was used with the ARF to obtain onsite and offsite doses as shown in Table 3-9. The estimated total dose onsite is 3.24 rem EDE and the estimated total dose offsite is 2.72×10^{-01} rem EDE. The onsite and offsite dose estimates were made without credit for the sprinkler system.

3.1.5 Risk Evaluation

Table 3-8 provides the dose consequence for a liquid spray release from the 340 Vault Tank while pumping to the railroad tanker. The conditions for the release are conservative in that no credit has been taken for the building enclosing the tanker, nor the loss in head pressure from the pump to the tanker; and it is estimated that an entire vault tank volume will be pumped without any awareness of the leak during this $2\frac{1}{2}$ hour operation. The onsite and offsite doses are based on a full tank (57 kL) of the worst-case source term (Table 3-8). Figure 3-1 shows a comparison of the total dose consequences to the risk-acceptance guidelines (*A Model for Estimating the Release Rate of Aerosol Droplets From Pressurized Liquid Leaks*, WHC 1992) for an Annual Frequency Category of 1.0×10^{-5} . Both the onsite and offsite doses are within the risk-acceptance guidelines.

The dose consequences for the worst-case solid waste fire in the 340-B West Building is shown in Table 3-9. Conservatism has also been applied here, by calculating the entire inventory of steel drums and fire retardant boxes to be consumed in the fire and taking no credit for the building or sprinkler system. The onsite and offsite doses are based on the worst-case source term (Table 2-4). Figure 3-1 shows a comparison of the total dose consequences to the risk-acceptance guidelines for an Annual Frequency Category of 1.0×10^{-3} . Both the onsite and offsite doses are below the risk-acceptance guidelines.

The release fraction uncertainties can be obtained from DOE-HDBK-3010-94. This ISB document uses the bounding release fractions for the dose estimates. For a solid waste fire the bounding release fraction is $5 \text{ E-}04$ and the median release fraction is $8 \text{ E-}05$ (DOE-HDBK-3010-94, p. 5-1). The expected onsite and offsite doses, using the median release fraction, is, respectively, 0.518 rem and 0.0435 rem (verses the bounding values of 3.24 rem and 0.272 rem, respectively). The bounding values, therefore, are 6.25 times higher than the expected values and this difference represents the uncertainties for the consequences. The uncertainty for the frequency value, $1 \text{ E-}03$, is the range $1 \text{ E-}04$ to $1 \text{ E-}02$. The fire accident was judged to be an unlikely event with a frequency range of $1 \text{ E-}04$ to $1 \text{ E-}02$. The value $1 \text{ E-}03$ is in the middle and was chosen for Figure 3-1 to show the comparison to the guideline values.

The above calculations show that neither a spray release in 340-B East Building, nor a solid waste fire in 340-B West Building would exceed the 5 rem EDE onsite or the 0.5 rem EDE offsite limits. The 340 Facility continuously receives and transports waste and thus the inventory may change. To ensure that onsite/offsite limits are not exceeded during a spray-release incident, the RLWS inventory must be known to be within the bounding source term (see 3.2.2.3.2) before loadout operations are initiated. Also, the inventory of solid wastes must be controlled within the solid waste bounding source term (see 3.2.2.3.3).

3.2 HAZARD CONTROLS

This section describes the physical and administrative systems used for the control and/or mitigation of the 340 Facility accidents and events. There are no Safety Class nor Safety Significant systems in the 340 Facility.

3.2.1 Structures, Systems and Components (SSC)

3.2.1.1 340 Building. As discussed in section 2.3.1, the 340 Building is an above ground structure of steel and sheet metal attached to a steel frame mounted on a concrete foundation. The 340 Vault is constructed of concrete with cover blocks, access holes, and hatches.

The 340 Building truck lock, control room, and decontamination room are protected with smoke and temperature detectors and automatic sprinklers. The smoke detectors and the sprinkler systems alarm locally and at the central dispatch in the 200 Area. Criteria, as established by the National Fire Protection Association Codes, were used for location of the above systems.

The 340 Building is provided with a ventilation supply system, an evaporative cooling system, an electric heating system, and an exhaust system servicing all of the rooms except the control and mechanical equipment rooms. The exhaust system consists of two-stage HEPA filters, an exhaust fan, and a stack that is located above the rest rooms. Airflow from the decontamination area hood, sink, and sump pass through an additional stage of high efficiency filters before the two-stage HEPA filters.

The 340 Vault and Tanks are serviced by a common exhaust system, discharging through a stage of dust stop filters, two-stage HEPA filters, and two exhaust fans located just east of the vault. The capability exists for the addition of activated charcoal adsorbent into the exhaust treatment train. The charcoal filters were used in the past to help control potential I-131 releases associated with activities in the 327 Postirradiation Testing Laboratory. Some ventilation alarms annunciate locally at the stack and remotely through the alarm horns in the 340 control room and through the 310 TEDF plant alarm horns.

A process radiation detector is located in Valve Box 9. It provides local alarms and remote alarms at the 340 Building Control Room Annunciator Panel and the 310 TEDF control room. The detector provides a means of monitoring for a change in the radiation level or movement of radioactive liquids and also provides an alarm on high radiation levels.

Vault tank high liquid-level alarms exist that aid in preventing tank overflow. There are also influent alarms that will alarm before the high-level alarms (variable-level alarms are normally set slightly above the current level in the tank). The tanks are also equipped with a high-high set point that had been used to automatically pump the waste to the 340-A Tanks. These alarm remotely in the 340 control room and in 310 TEDF. Finally, there is an overflow leg between the two vault tanks so they will overflow to each other. In the event of a tank or piping leak, the concrete vault will temporarily contain up to 360 kL (94,000 gal) of liquid. The vault is equipped with a leak detector located at the sump and a sump pump, although the pump is de-energized as part of the vault tank isolation.

3.2.1.2 340-A Building. The 340-A Building is constructed of steel panels attached to a steel frame mounted on a concrete foundation. The floor is set about 1 m below grade level. The floor and lower portions of the walls are made of concrete. The floor contains a dry sump that used to drain back to the 340 Vault Tanks. Six 30 kL (8,000 gal) stainless steel tanks are housed in the building to provide reserve capacity for RLW. A leak detector is located in the trench to the sump. The tanks are ventilated through the vault ventilation system.

3.2.1.3 340-B Building. The 340-B Building is constructed of steel panels mounted on a concrete foundation. The building is divided down the center by a 30 cm (12 in.) thick by about 3.7 m (12 ft.) high concrete shielding wall, topped by a standard 4 inch stud/rock wall to the ridge.

The east side of the building is serviced by a ventilation system located in the west side of the building, consisting of a pre-filter, HEPA filter, exhaust fan, and stack; operated only when needed (see 2.3.3).

The 340-B Building is equipped with automatic heat activated sprinklers. Installation locations were established by guidelines in the National Fire Protection Codes.

3.2.1.4 340 Facility Compliance with Codes, Standards, and Regulations.

The core 340 Facility was built and commissioned for radioactive service in 1952, with several additions acquired over the years. The facility met the applicable codes, standards, and regulations at the time of construction. The facility has been upgraded to meet changing emission and regulatory requirements. The facility and operations are routinely audited to ensure compliance with the intent of meeting applicable codes, standards, and regulations. As part of the overall 300 Area upgrade, the RLWS has been renovated to improve environmental protection and to further reduce potential for personnel exposure. Although not required to mitigate, prevent, or control design base accidents, the systems listed in Table 3-10 provide enhanced safety in the operation under both routine and abnormal situations.

3.2.2 Administrative Controls/Operational Restrictions

3.2.2.1 Administrative Controls. The 340 Facility operations are conducted in accordance with various operational and administrative documents that establish formal programs to ensure compliance with applicable DOE Orders. Where necessary, instructions and procedures are prepared to translate the general requirements of DOE and PHMC directives into specific guidance and limits to be observed by the operating staff.

3.2.2.2 General Project Hanford Management Contract Administrative Controls.

Hanford policy and standard practices regarding specific facets of ongoing operational activities are set forth in procedures approved for implementation by the PHMC. Several of the more significant documents that constitute part of the PHMC system for administrative controls are included in the following:

- DOE/RL-96-109, Rev. 0, *Hanford Radiological Control Manual*
- HSRM-1, Rev. 2, *Hanford Site Radiological Control Manual*
- HNF-IP-0263-340, *Building Emergency Plan*
- HNF-IP-1000, *300 Area Liquid Effluent Facilities Administration*
- HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*

Project Hanford Procedure Topic Listings

- *Emergency Management*
- *Occupational Safety & Health*
- *Transportation and Packaging*
- *Quality Assurance*
- *Fire Protection*
- *Safeguards & Security*
- *Waste Management*
- *Engineering Program*
- *Environmental Protection*

3.2.2.2.1 340 Facility Waste Management Facility Administrative Controls. Operation of the 340 Facility is conducted in accordance with operating, maintenance, and administrative procedures and programs approved by WMH management, WMH-100, *Waste Management Hanford Policies* and WMH-200, *Waste Management Hanford Procedures*. These procedures and programs are incorporated into 340 Facility operating procedures and the HNF-IP-1000, *300 Area Liquid Effluent Facilities Administration*. These policies and procedures are consistent with and implements the requirements of the documents listed above. Any modification that would put the facility outside the boundary of the safety analysis would be an unreviewed safety question and would be documented in subsequent revisions or supplements to this ISB.

Specific procedural and administrative controls are provided that cover:

- Building emergency preparedness and response procedures
- Equipment and process operating procedures
- Maintenance and inspection procedures
- Staff qualification and training programs — includes training for building personnel in the areas of criticality safety, radiation protection, industrial safety, job hazards, security, operating and emergency procedures.

These controls are implemented at the work stations (point of operation) by:

- Qualifications and demonstrated proficiency for job assignment
- The 340 Facility procedures and work packages that contain the task-specific Job Safety Analysis and Radiological Work Permits.

3.2.2.2.2 Assessment Program. Adherence to established procedures and the adequacy of those procedures is verified continuously by a program of audits and inspections.

The audit program is accomplished by the following:

Independent Assessment — Independent assessments in compliance with 10CFR830.120(c)(3)(ii). "Independent Assessments," and DOE Order 5700.6C(9)(b)(3)(b), "Criterion 10-Independent Assessment," are the responsibility of FDH in accordance with HNF-MP-599, *Project Hanford Quality Assurance Program Description*, Section 10, "Independent Assessment."

Management Assessment — WMH assessments are performed periodically by personnel familiar with the subject being assessed. Management will review and analyze assessment results for compliance issues, conditions adverse to quality, opportunities for quality improvement, including lessons learned, and applicability to their areas of functional responsibility. Assessments are performed in accordance with WMH-200, Section 1.2, "Assessment Program."

Safety — Safety is responsible for conducting regular audits to verify the adequacy of the operational procedures, records, and conformance with operating limits.

Quality System Assessments — Quality assurance assessments will be performed periodically by personnel familiar with the subject being assessed, to verify compliance with the requirements of HNF-SD-WM-QAPP-036, "Waste Management Federal Services of Hanford, Inc., Quality Assurance Program Plan".

3.2.2.3 Operational Restrictions. This section identifies the interim operational safety requirements (IOSR) for the 340 Facility. The IOSR are administrative controls that define limits to assure compliance within the classification of a Hazard Category 3 facility. These limits replace the previous operating limits.

3.2.2.3.1 340 Facility Maximum Fissile Limit.

Applicability — All storage, handling, or processing of radioactive waste in the 340-Complex.

Objective — To maintain a fissile materials inventory compliant with the classification of an isolated facility per HNF-PRO-334 and the *CSER 94-012: Criticality Safety Evaluation Report for 340 Facility* (Miller, E.M., 1996) which supports the isolated facility classification.

Requirement — The ^{239}Pu FGE equivalent quantities within the 340 Facility will be maintained at less than 1/3 of a minimum critical mass (MCM).

Surveillance — An inventory of the FGE within the 340 Facility will be maintained to demonstrate compliance with the requirement. The administrative program assigns responsibility and establishes programs for sampling and surveillances which verify adequate inventory accountability.

Basis — The facility criticality safety status as an isolated facility is in accordance with HNF-PRO-334. The corresponding maximum fissile inventory value is based upon a fraction of the minimum critical mass (MCM). This MCM is in turn based upon 100% Pu 239 aqueous metal solution, optimum water moderation, full water reflection, optimum concentration (31g Pu/L) and spherical geometry. By contrast, the large, cylindrical 340 tanks contain much lower fissile concentrations, along with other chemical compounds which serve to absorb neutrons (e.g., chlorides, nitrates, and nitrites).

Recovery — In the event the requirement is not met, notification will be immediate and will include the 340 Facility manager and notifications as required according to HNF-PRO-549, *Criticality Safety Nonconformance Response*. The cause of the event will be determined and a course of action to correct the problem will be implemented. A determination will be made if a continued classification as a Hazard Category 3 facility is valid.

Audit Point — Sample data used to determine FGE amounts and inventory lists of FGE amounts will be retained for a minimum of 15 months.

3.2.2.3.2 Radioactive Solid Waste Storage Limit.

Applicability — All storage, handling, or processing of solid radioactive waste in the 340-Complex subject to fire.

Objective — To prevent significant onsite and offsite dose consequences in the event of a fire by maintaining the solid radioactive materials inventory at a level validating the solid waste storage areas as Hazard Category 3.

Requirements — Dispersible radioactive materials packaged in solid waste in fire retardant boxes or drums will not exceed the following limits:

- Alpha emitters 1.5 Ci
- Beta-Gamma emitters 10 Ci.

Surveillance — An inventory of the radioactive materials present will be maintained that demonstrates compliance with the requirement.

Basis — The accident analysis for a fire involving radioactive waste materials (Section 3.0) with the above maximum inventory has shown that the resultant consequences, without credit for mitigating features, meets the criteria for a Hazard Category 3 facility. The analysis shows that the estimated fire consequences for a total solid waste inventory of 11.5 Ci would be 3.24 rem EDE onsite and 2.72×10^{-1} rem EDE offsite.

Recovery — In the event the requirement is not met, notification will be immediate and will include the 340 Facility manager. The cause of the event will be determined and a course of action to correct the problem will be implemented. A determination will be made if a continued classification as a Hazard Category 3 facility is valid.

In the event that the inventory is not maintained, the facility inventory of radioactive material will be immediately determined and corrective action taken to prevent reoccurrence

Audit Point — Storage/disposal records of inventory lists of radioactive materials will be retained for a minimum of 15 months.

3.2.2.3.3 340-B Waste Loadout Restrictions.

Applicability — 340 Facility RLWS transfer operations to 340-B East Building for waste tanker loadout.

Objective — To prevent significant onsite and offsite consequences in the event of a spray leak by maintaining the liquid radioactive materials inventory at a level validating the vault tanks as Hazard Category 3.

Requirement — If the sending tank exceeds the following criteria, RLWS transfer operations to the 340-B East Building for waste tanker loadout are prohibited until Engineering provides isotopic calculations that verify the inventory to be transferred will not exceed the onsite/offsite doses and Hazard Category 3 Limits.

- Total Alpha 66 Ci
- Total Beta/Gamma 39 kCi.

Surveillance — Tank inventory data, by sample analysis and calculations, will demonstrate compliance with the requirement before transfer to a waste tanker.

Basis — The accident analysis for a spray release (Section 3.0) of radioactive liquid waste from the vault tanks with the above maximum inventory has demonstrated that the resultant onsite and offsite consequences, without credit for mitigating features, classify the 340 Facility as Hazard Category 3. The spray-release accident analysis shows that the estimated release consequences from a full vault tank inventory of 1.07 m^3 released (the respirable release is $5.11 \times 10^{-4} \text{ m}^3$) is 3.92 rem EDE onsite dose and 3.30×10^{-1} rem EDE offsite.

Recovery — In the event the requirement is not met, transfer operations will cease, notification will be immediate, and will include the 340 Facility manager. Restrictions on quantity transferred or isotopic calculations from engineering are required for continuation of transfer operations. The cause of the event will be determined and a course of action to correct the problem will be implemented.

Audit Point — Vault tank sample data and waste transfer records/memo will be retained for a minimum of 15 months.

3.2.3 Institutional Safety Programs

3.2.3.1 As Low As Reasonably Achievable. In compliance with DOE Orders and to support radiation protection, PHMC has established a company-wide ALARA program applicable to the pre-operational, operational, and post-operational phases of all projects and facilities. The program goal is to minimize human exposures (occupational and public) to radiological and nonradiological hazards and conditions ALARA, commensurate with sound economics and operating practices. The program focuses on those activities with significant risk because of the presence of radiation, toxic substances, or hazardous conditions.

The PHMC policy on radiation protection is described in the Hanford Project Procedures Topic Listing *Radiation Protection* and DOE/RL-96-109, Rev. 0, *Hanford Radiological Control Manual*. The WMH ALARA Program is described in WMH-200, Section 3.2, *ALARA Program*.

Responsibility for ensuring implementation of ALARA practices and principles is assigned to the PHMC Team ALARA Committee. The ALARA committee is responsible for coordinating administration of company-wide activities and the Facility ALARA Committee is established to implement ALARA practices and principles on an operational level. ALARA surveillances are conducted periodically by Radiological Engineering and the measures are implemented in day to day activities.

3.2.3.2 Operational Considerations. A number of administrative controls are in place ensuring safety during maintenance activities. In particular, HNF-PRO-069, *Maintenance Management* presents the PHMC policy on maintenance management and work control. WHM-200, Section 3.1, *Maintenance Work Management* and HNF-IP-1000, Section 4 provide for pre-job planning, a job hazards analysis, and the dual function of providing a responsible review of the work scope and zone, as well as heightening operator awareness of personal safety.

Operations and maintenance activities within the 340 Facility will be governed by technical procedures that incorporate ALARA and safety concerns. Maintenance activities are covered by preventive maintenance procedures.

In addition to safety guidance provided in operation and maintenance procedures, activities that involve radiological exposure must be performed in compliance with the protective measures identified in DOE/RL-96-109, Rev. 0, *Hanford Radiological Control Manual*. Entry into, and the work performed in any radiological area, is governed by the provisions of a radiological work permit (RWP) issued for that specific job or area. Radiological Control issues RWPs with the review and concurrence signature of Facility Operations management. The RWP must be followed by any person involved in the scope of operations described in the permit and only work specified in the RWP may be performed.

3.2.3.3 Area Radiation and Airborne Radioactivity Monitoring Instrumentation. Portable monitoring equipment may be used in support of the Radiation Protection Program developed for the 340 Facility operation to detect, record, and disseminate results of radiation and contamination surveys conducted in and around the facility. Monitoring may be performed before operations or maintenance activities involving surface contaminated areas.

Air sampling will be performed as required by the particular operations being conducted according to the criteria in HSRCM-1.

3.2.3.4 Radiological Control Program. This section provides information on the PHMC Radiological Control Program organization, procedures, equipment, instrumentation, and facilities.

3.2.3.4.1 Radiological Control Organizations. The PHMC Radiation Protection Program (RPP) commits the PHMC to implement the essential elements of an acceptable ALARA program from the *Implementation Guide* (IG) for 10CFR835 titled *Occupational ALARA Program*. The RPP cites the IG as an implementing provision and states that "FDH will implement this requirement (835.101(c)) through the FDH Occupational ALARA Program." The second essential element (Assignment of Responsibilities) of an acceptable ALARA program requires the assignment of specific responsibilities for implementing the ALARA program. The procedure, HNF-PRO-1619, *ALARA Organization and Responsibilities*, assigns many responsibilities, but it does not assign all ALARA responsibilities. The balance of specific ALARA responsibilities not assigned by HNF-PRO-1619 are assigned by facility specific ALARA programs.

The *Waste Management Hanford Procedure*, WMH-200, Section 3.2, "ALARA Program" states the following:

"Each nuclear or radiological facility and all radiological work performed under WMH contract must meet the ALARA work planning processes described in WMH-200, Section 3.8, "Enhanced Radiological Work Planning," and the following 11 essential elements of an ALARA Program, as appropriate based on the facility/project risk level. Radiological Control is responsible for the following:

1. Management commitment
2. Assignment of responsibility
3. Administrative control levels
4. Radiological performance goals
5. ALARA training
6. Plans and procedures
7. Internal audits and assessments
8. Optimization methodology (decision-making methods)
9. Radiological design reviews
10. Radiological work/experiment planning
11. Records"

WMH-200, Section 3.2, *ALARA Program* also states:

"WMH facility/project/organization ALARA committees shall perform assessments of the ALARA process no less frequently than every 3 years to identify strengths and weaknesses in the program. The assessments should be conducted by individuals not directly responsible for implementing the facility/project/organization ALARA Program being assessed."

The Facility Evaluation Board is responsible for conducting facility assessments in accordance with established criteria and schedules to verify that operational and management processes are in place which ensure compliance with DOE Order 5482.1B, *Environmental, Safety and Health Appraisal Program*. Examples of activities covered by this organization include readiness reviews, design reviews, audits and appraisals, facility inspections, and operational surveillance assessments.

HPTs are trained and certified in accordance with 10 CFR 835, *Occupational Radiation Protection*, Subpart J, and are responsible for the day-to-day field implementation of the radiation protection program.

3.2.3.4.2 Equipment, Instrumentation, and Facilities. The following sections provide information on the various radiological control-related equipment, instrumentation, and facilities available to support the radiation protection program for the 340 Facility. The design, installation, operation, and calibration of radiological control instrumentation will be in accordance with WHC-SD-GN-DGS-30011, *Radiological Design Guide*, DOE/RL-96-109, and WHC-IP-0718, *Health Physics Procedures Manual* (WHC 1991).

Air-Sampling Instruments — Stationary air-sampling instrumentation are in place in the 340 Building stack, the 340-B Loadout Facility stack, and the 340 vault stack. Radiological Control may use portable air samplers. Gooseneck and high-volume grab samplers are used to provide long- and short-term sampling of areas that have the potential to exceed 10% of a Derived Airborne Concentration (DAC) during certain job evolutions.

Personnel Survey Instruments — Semiportable rate meters and PCM-IBs for monitoring personnel are available at the 340 Facility. Both scintillation and Geiger-Mueller detectors are available for use by 340 Facility personnel.

3.2.3.4.3 Procedures. PHMC/WMH has compiled documents grouped into Topic Listings that describe requirements and procedures relative to radiological and environmental protection. A brief overview of relevant Topic Listings and summaries of various radiological protection procedures applicable to 340 Facility activities are provided below.

- *Radiation Protection*, provides methods, routine practices, controls, exposure guides, supporting data, and other information developed to assist radiological control personnel in establishing and maintaining a uniform and sound radiation control program.
- *Environmental Protection*, provides detailed standards for controlling the release of radioactive and nonradioactive materials to air, water, and land, environmental surveillance criteria, and effluent sampling and monitoring program requirements.

Radiological Control Surveys — The methods, frequencies, and procedures for conducting radiation surveys are determined based on the characteristics of materials handled (i.e., form, element, isotope) and the radiological condition and type of operations conducted (i.e., dose rate, contamination level, or airborne potential).

Contamination Control — Bases and methods to control contamination are discussed in DOE/RL-96-109. Actions to be taken if defined limits of contamination control are exceeded address personnel contamination and subsequent decontamination and documentation, along with control of equipment and surfaces that exhibit contamination. The basic policy is to preclude contamination through facility design by including mechanical constraints such as air filtration and temporary maintenance enclosures.

Safety Training — Safety training will be provided to all employees who work at the 340 Facility. In addition to radiation protection training, the PHMC *Training & Qualification* Topic Listing consists of courses in general safety awareness, nuclear safety, hazardous materials, and waste. All managers are responsible for properly training their personnel to perform assigned tasks in a manner that minimizes risk to themselves, co-workers, and the public; that minimizes negative impacts to the environment; that minimizes risk of damage to plant and equipment; and that demonstrates competent and cost-effective performance. The Topic Listing titled *Training and Qualification* describes procedures for course development and maintenance, scheduling, instructor certification, training delivery, student evaluation, and maintenance of training records.

The Project Hanford Procedure, HNF-PRO-082, *Radiological Worker Training*, includes a multi-tiered radiation protection training program that provides a level of training commensurate with an individual's classification as a radiation worker/nonradiation worker, as well as the individual's specific work assignment. The training includes General Employee Radiological Training, Radiological Worker I and II Training and Radiological Worker Refresher Training. Requirements for radiation protection training are stated in DOE/RL-96-109 and are consistent with the requirements stated in DOE N 441.1 and 10 CFR 835.

Personnel Exposure Monitoring — PHMC dosimetry program establishes requirements for measuring and recording personnel exposure. This program is described in DOE/RL-96-109, Rev. 0, *Hanford Radiological Control Manual*. A summary of these requirements is provided in this subsection.

The PHMC requires that all employees, visitors, and vendors be issued and use an assigned record dosimeter at all times, when in radiological areas within a facility. These dosimeters provide an official measurement of an individual's radiation exposure. All record dosimeters used by PHMC personnel are accredited by the DOE Laboratory Accreditation Program. In addition to the Hanford Standard multi-purpose dosimeters, radiation workers may be required to wear supplemental dosimetry to monitor highly-localized doses. These devices may include finger rings, pencils, electronic dosimeters, two-chip dosimeters, wrist dosimeters, or special beta-photon dosimeters.

Exposure to internally deposited radionuclides is assessed by in vitro (excreta analysis) or in vivo (whole-body or chest-count) bioassay measurements. A baseline bioassay determination of internal radionuclides is performed for employees, visitors, and vendors as a prerequisite to any work assignment that involves the potential for internal deposition of radioactive material. Based on a review of the radionuclides and activity levels involved, the WHC Dosimetry Program determines the appropriate type of surveillance and establishes a routine internal exposure surveillance schedule.

As determined by Nuclear Safety, based on HNF-PRO-546 and 340 CSER criteria, there is no need for criticality detection in the 340 Facility areas.

Occupational Radiation Exposure Records System — DOE/RL-96-109, *Hanford Radiological Control Manual* describes a system that maintains complete, accurate, and private personnel exposure data. The data include whole-body dose (neutron, beta, and gamma), extremity-dose (beta and gamma to fingers), and internal depositions. Hanford Site employees will have whole-body doses measured at a frequency dependent upon their work assignments as specified under DOE/RL-96-109.

Airborne Radioactivity Controls — Detailed methods and procedures for evaluating and controlling airborne radioactivity are discussed in WHC-SD-GN-DGS-30011. Radiological Control may employ portable instrumentation to evaluate airborne radioactivity concentrations. Alpha, beta, and gamma-emitters in air samples, as well as plutonium, noble gases, and iodine may be measured. Samples will be analyzed routinely, and the results may be reported to the Radiological Control, Facility Operations, Radiological Engineering, and ALARA organizations.

Airborne radioactivity concentrations will be controlled by using the design requirements stated in WHC-SD-GN-DGS-30011. Respiratory protection equipment may be required under special circumstances, as described in DOE/RL-96-109.

3.3 SUMMARY

The 340 Facility RLWS and solid waste operations, with the controls outlined in this document, pose an acceptable risk to both the onsite and offsite individual. These operations maintain a radionuclide and chemical inventory without reliance on Safety Class or Safety Significant air-confinement barriers to reduce the amount of release to the onsite and offsite individual.

In accordance with DOE-STD-1027-92, the 340 Facility has been determined to be a Hazard Category 3 Facility. The credible accident scenarios, with a frequency greater than 10^{-6} per year, are an RLWS spray leak while pumping from the 340 Vault Tanks to a waste tanker and a solid waste fire in 340-B West Building. Interim operational safety requirements have been developed to assure that the facility is maintained as a Hazard Category 3. Controlling the radioactive and hazardous material inventories ensures that potential onsite and offsite accident consequences are within the risk-acceptance guidelines.

Figure 3-1. Onsite and Offsite Radiological Risk Comparison to Guidelines.

[Note: 10 mSv = 1 rem.]

Table 3-1. 340 Vault Tank Hazard Category Calculations.¹

Isotope ²	Facility inventory (Ci/L) ³	Ci/tank 57 kL (15,000 gal)	Category 2 TQ (Ci)	Modified Category 2 TQ _m (Ci) ⁴	Ci/TQ (Ci) ^m
Beta/gamma	6.90 E-01	3.93 E+04	2.20 E+04	2.47 E+06	1.59 E-02
Alpha	1.16 E-03	6.61 E+01	5.60 E+01	6.18 E+03	1.07 E-02
			Sum of ratios	Total	2.66 E-02

¹Based on the bounding source term in Table 2-3 and the spray-release accident.²³⁹Pu.
²For a worst-case inventory beta/gamma is taken as ⁹⁰Sr and alpha is taken as ²³⁹Pu.
³Simplified inventory based on total Ci/L inventory for beta/gamma and alpha emitters.
⁴The Category 2 threshold quantities were modified as shown in Chapter 3.0
 TQ = threshold quantities.
 TQ_m = threshold quantities modified.

Table 3-2. Hazard Categorization of 340 B Building Solid Radioactive Waste.

Isotope	Worst-case inventory of radioactive solid waste (Ci)	Category 2 TQ (Ci)	Total Ci/TQ (Ci)
Beta/gamma	10.00 E+00	2.20 E+04	4.55 E-04
Alpha	1.50 E+00	5.60 E+01	2.68 E-02
		Sum of ratios	2.72 E-02

NOTE: For a worst-case inventory beta/gamma is taken as ⁹⁰Sr and alpha is taken as ²³⁹Pu.
 TQ = threshold quantities.

Table 3-3. Calculation of Plutonium Equivalent Curie/Liter.¹

Isotope	(Ci/L)	Correction ² factor	PE (Ci/L)	Specific ³ activity (Ci/g)	(g/L)	Fraction of total
²⁴¹ Am	6.26 E-04	1	6.262 E-04	3.42 E+00	2.08 E+01	0.1134108
²⁴⁰ Pu	7.85 E-05	1	7.85 E-05	2.27 E-01	3.93 E+01	0.2143256
²³⁹ Pu	5.81 E-05	1	5.81 E-05	6.20 E-02	1.06 E+02	0.5804136
²⁴² Pu	2.35 E-07	1/1.1	2.43 E-07	3.93 E-03	6.78 E+00	0.0370055
²³⁸ Pu	2.72 E-04	1/1.1	2.47 E-04	1.71 E+01	1.81 E+00	0.0098606
^{243/244} Cm	1.67 E-06	1/1.9	8.77 E-07	8.10 E+01	2.34 E-03	0.0000127
²⁴¹ Pu	7.48 E-03	1/52	1.44 E-04	1.03 E+02	8.24 E+00	0.0449712
Total			1.16 E-03			1.0

¹Based on Bounding Source Term (Table 2-1) and Appendix C of WMC-EP-0063-4.
²WMC-EP-0063-4, Table C-2.
³WMC-EP-0063-4, Table C-7.

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (7 sheets)

Hazard/ energy source	Potential accident and sequence	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
I. Chemistry						
A. Fire						
1. Range	None.	None.	Area around buildings is paved and/or graveled.	--	Incredible	IV
2. Building						
2a. 340	Fire could start in office area.	Fire and smoke damage, personnel injury.	Fire detection and suppression system.	--	Unlikely	III
2b. 340 Decontamination	Fire could start in or spread to stored dry waste.	Radiological/ Toxicological release could occur locally, onsite and offsite.	Fire detection and suppression system.	--	Unlikely	III
2c. 340 Control Room	Fire could start in wiring.	Radiological/ Toxicological contamination could occur locally.	Fire detection and suppression system.	--	Unlikely	III
2d. 340-A	Electrical fire could occur.	None.	Metal building, metal tanks, and piping.	No flammable or combustible material stored in building.	Extremely Unlikely	III
2e. 340-B East Building	Fire could start in wiring.	Radiological/ Toxicological contamination could occur locally.	Metal building, metal tanks, and piping. Lighting and pumps only power source. Fire detection and automatic sprinkler system.	No flammable or combustible material stored in building.	Extremely unlikely	III
2f. 340-B West Dry Waste Storage	Fire could start in stored waste.	Radiological/ Toxicological release could occur locally, onsite and offsite.	Fire Detection and automatic sprinkler system.	--	Unlikely	III
3. Liquid Petroleum Gas						
3a. Fork lift or JLG manlift LPG tank	Tank damage causes leak. Resulting fire impacts stored waste.	Radiological/ Toxicological release could occur locally, onsite and offsite.	Design of fork lift and JLG protects tank from damage. Fire detection systems	Training of operators in proper handling techniques.	Anticipated	III
3b. LPG tank near 340 bldg. for auxiliary heater	Damage to LPG tank causes leak, resulting fire impacts stored waste.	Radiological/ Toxicological release could occur locally, onsite and offsite	Design of tank storage cage protects tank. Fire suppression system in 340.	Flammable material not stored or permitted near LPG tanks.	Anticipated	III

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (7 sheets)

Hazard/ energy source	Potential accident and sequence	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
I. Chemistry (Continued)						
3c. Any LPG tank.	External fire superheats tank resulting in BLEVE.	Radiological/ Toxicological release could occur locally, onsite and offsite.	(see A.3a & b) Also, LPG tanks have DOT mandated pressure relief valves.	Absence of fuel source for heating tank.	Incredible	II
4. Natural gas pipeline						
4.a Excavation equipment	Pipeline punctured and gas ignited	Fire/explosion, flame jet, personnel injury	Above ground markers identify pipeline location Nearest pipeline >12.5 m from 340 Facility	Excavation permit required, hand digging required near pipelines	Extremely unlikely	III
4.b Pipeline deterioration	Pipeline leaks	Personnel asphyxiated	Pipeline materials are corrosion resistant Gas leak would be diluted by outside air Gas odorant alerts personnel Nearest pipeline >12.5 m from 340 Facility	--	Extremely unlikely	III
4.c Packaged steam boiler	Gas leak, flammable gas-air mixture, ignition source	Fire/explosion, missiles generated, shock wave, personnel injury	Boiler controls prevent explosive conditions Boilers housed in noncombustible annex buildings Nearest boiler annex >25 m from the 340 Facility	Training of boiler operators	Extremely unlikely	III
B. Corrosion						
1. Corrosion fails drum/container						
1a. Dry Waste	Dry waste released to air.	Radiological/ Toxicological contamination could occur locally.	--	No long-term storage of waste drums.	Extremely unlikely	IV
1b. Liquid waste drums	Spill occurs.	Radiological/ Toxicological contamination could occur locally.	Area has gravel around building that reduces evaporation. Area inside the building is contained.		Unlikely	III

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (7 sheets)

Hazard/ energy source	Potential accident and sequence	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
I. Chemistry (Continued)						
2. Corrosion fails tanks or piping						
2a. 340 Vault Tanks	Corrosion in tank or piping allows release.	Release to ground could occur.	Tanks and most piping under ground. Pipes and tank double encased. Tank vault is concrete. Leak detection system.	--	Unlikely	III
2b. 340-A Tanks	Corrosion in tank or piping allows release.	Release to ground could occur.	Pipes double encased. Concrete base and sump. Leak detection system.	--	Unlikely	III
II. Potential/kinetic/energy						
A. Equipment						
1. Heavy equipment impacts drums/containers	Drums or boxes ruptured.	Radiological/ Toxicological contamination could occur locally.	--	Only certified, experienced, operators, are allowed in facilities.	Unlikely	III
2. Sharp/hard equipment impacts drums/containers	Drums or boxes ruptured.	Radiological/ Toxicological contamination could occur locally.	--	Only certified, experienced, operators, are allowed in facilities.	Unlikely	III
3. Drum/container dropped	Container splits open.	Radiological/ Toxicological contamination could occur locally.	Some drums lined with ten mil nylon reinforced plastic.	Drums can take 3 m drop without failing.	Extremely Unlikely	III
4. Liquid waste drum dropped or damaged	Liquid spill to ground.	Ground contamination could occur.	Ground around building covered with gravel. Inside areas are contained.	--	Unlikely	III
5. Pipe or pipe connection fails while pumping liquid waste to tankcar	Spray release could occur.	Radiological/ Toxicological release could occur locally, onsite and offsite.	HVAC system operating during pumpout - radiation alarms.	Piping inspected prior to connection. Leak check at start of pumping. Periodic inspections for leaks.	Extremely unlikely	III
6. Ventilation system, fails to maintain a vacuum in the vault/tanks.	Vault/tank pressurizes above atmospheric pressure.	Radiological/ Toxicological contamination could occur locally.	Ground around building covered with gravel.	Procedures provide recovery actions for operations.	Unlikely	III

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (7 sheets)

Hazard/ energy source	Potential accident and sequence	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
II. Potential/kinetic/energy (Continued)						
7. Ventilation duct insulation deteriorates.	Internal ice buildup during freezing weather, vacuum not maintained.	Radiological/ Toxicological contamination could occur locally.	Ground around building covered with gravel.	Periodic inspection and maintenance or replace worn insulation	Unlikely	III
B. Helicopter/or plane crash	Crash into facility occurs.	Radiological/ Toxicological release could occur locally, onsite and offsite.	--	Flights over facility are rare.	Incredible	II
C. Transport Vehicle	Waste drums dropped while loading or unloading.	Radiological/ Toxicological contamination could occur locally.	--	Only certified, experienced, operators, are allowed in facilities. Drums can take 3 m drop without failing.	Unlikely	III
III. Natural occurrences						
A. Flood	The probable maximum flood could produce a river elevation of 116 m (382 ft) above msl which is 1.5 m (5 ft) below ground level at the 340 Building 118 m (387 ft). The floor level of the 340 Vault is at 111 m (365 ft) above msl.					
1. High water	High river flood covers area.	Overflow of vault tanks to river.	--	--	Extremely unlikely	III
2. Heavy rain (flash flood)	Heavy rain inundates area.	Overflow of vault tanks to river.	--	--	Unlikely	III
B. Seismic event	Strong earthquake collapses buildings ruptures tanks					
1. 340	Office area destroyed.	Property damage and personnel injury could occur.	--	--	Unlikely	IV
2. 340 Decontamination	Sample and packaging area destroyed.	Minor release could occur.	--	--	Unlikely	III
3. 340 Control Room	Instrument panel and Controls destroyed.	Minor release could occur.	--	--	Unlikely	III
4. 340-A	Tanks ruptured concrete base cracked.	Ground release could occur.	--	Current activities are only related to monitoring and clean-out.	Unlikely	III
5a. 340-B East receiving (empty)	Loading platform and pumping equipment damaged or destroyed.	Minor radiological/ toxicological release could occur locally.	--	--	Unlikely	III

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (7 sheets)

Hazard/ energy source	Potential accident and sequence	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
III. Natural occurrences (Continued)						
5b. 340-B East (car in place)	Loading platform and pumping equipment damaged or destroyed, railroad car damaged.	Radiological/ toxicological release could occur locally, onsite and offsite.	--	--	Incredible	II
6. 340-B dry waste storage	Drums and boxes split open.	Radiological/ toxicological release could occur locally, onsite and offsite.	--	--	Extremely unlikely	III
C. Wind						
1. Missile	Wind blown missile could penetrate building.	Property damage and personnel injury could occur. Minor release possible.	--	Area kept free of projectiles.	Unlikely	III
2. Pressure	Building could be destroyed.	Property damage and personnel injury could occur. Minor release possible.	UBC, Zone II, which provides for a wind loading of 98 $\frac{\text{kg}}{\text{m}^2}$ (20 $\frac{\text{lb}}{\text{ft}^2}$) 142 Km/h (88 m/hr)	--	Unlikely	III
D. Volcanic Activity						
1. Ash buildup collapses building.	Building could be destroyed.	Property damage and personnel injury could occur. Minor release possible.	--	--	Unlikely	III
E. Lightning						
1. Lightning strikes building.	Electrical shorts and fire started.	Radiological/ toxicological release could occur locally, onsite and offsite.	All buildings on grounding grid.	--	Extremely unlikely	III
IV. Stack Release						
A. HEPA filters	Fire or HEPA filter failure causes stack release.	Minor Radiological/ toxicological release could occur.	DP gauge across filters.	Replacement on reaching set DP limit.	Extremely Unlikely	III

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (7 sheets)

Hazard/ energy source	Potential accident and sequence	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
V. Loss of Utilities						
A. Electrical	Storm or accident cuts off all electrical power.	Loss of stack flows and monitoring indication alarms and lights.	Emergency lighting.	Suspension of all waste handling operations.	Anticipated	IV
B. Process water	Loss of pump seal water.	Pump motor damaged by overheating.	Pump shutdown interlock.	--	Anticipated	IV
VI. Radiation						
A. Contamination	Marked radiation zones.	Persons working in zone could become contaminated.	Zones marked and/or isolated by rope fence.	ALARA rules are in effect. Personal Protective Equipment required.	Anticipated	III
B. Exposure	High-level radiation inventory	Personnel could receive a radiation dose.	Concrete cover blocks, and lead shielding.	--	Anticipated	III
C. Criticality	Fissile materials are present.	criticality could occur	--	Facility compliance with WHC-CM-4-29 (converted to Nuclear and Safety Systems Topic Listing).	Incredible	II
D. Hydrogen	Radiolysis of water generates hydrogen, the hydrogen accumulates and is ignited.	Property damage, personnel injury, and radiation exposure could occur.	Tank ventilation air flow dilutes hydrogen below flammable concentrations Backup exhaust blower Hydrogen generation rate is low No ignition source in tank or process vent system	Procedures provide for recovery actions on loss of vent flow.	Unlikely	III

ALARA = as low as reasonably achievable.

BLEVE = boiling liquid expanding vapor explosion

DOT = Department of Transportation

DP = differential pressure.

FNA = Fire Hazard Analysis.

HEPA = high-efficiency particulate air (filter).

HVAC = heating, ventilation, and air conditioning.

LPG = liquid petroleum gas.

msl = mean sea level.

UBC = Uniform Building Code.

Table 3-5. Probability Category Definition and Qualitative Accident Severity Levels.

Probability category definition			Qualitative accident severity levels	
Annual frequency category (AFC)	Category description	Nominal range of annual frequency	Severity category (SC)	Consequences to the public, workers, or environment
Anticipated (A)	An offnormal condition that individually may be expected to occur once or more during plant lifetime.	10^{-2} to 1	Category I	May cause deaths onsite or loss of the facility/operation, major injuries or illnesses offsite, radiation exposure to offsite individuals to individuals in excess of annual limits, or severe impact on the environment.
Unlikely (U)	Individually, condition is not expected to occur during the plant lifetime, but collectively, events in this category may occur several times.	10^{-4} to 10^{-2}	Category II	May cause severe injuries or severe occupational illnesses onsite, exposure to onsite individuals in excess of annual limits, major damage to a facility operation, minor illnesses or injury offsite, exposure to offsite individuals to radiation below annual limits, or major impact on the environment.
Extremely unlikely (EU)	Extremely low-probability conditions that are not expected during the plant lifetime, but that represent extreme or limiting cases of faults identified as possible. This category includes design basis accidents.	10^{-6} to 10^{-4}	Category III	May cause minor injury or minor occupational illness onsite, or exposure of onsite individuals to radiation below annual limits, negligible impact offsite, or minor impact on the environment.
Incredible (I)	Accidents for which no creditable scenario can be identified.	$<10^{-6}$	Category IV	Will not result in injury, occupational illness, exposure onsite or offsite, or result in a significant impact to the worker.

Table 3-6. Operating Hazards. (2 sheets)

Room location	Equipment	Hazardous material	Hazards							Mitigating features	Administrative controls
			Fire	Electric shock	Air borne	Noise	Surface contamination	Skin contamination	Whole body dose		
340 Building Vault	RLW vault tanks	RLW	--	--	X	--	X	X	X	Tanks below ground level. Concrete walls and .3 m 12" thick concrete shielded cover blocks. Access hatch, covered with penthouse and locked.	Entry for maintenance and inspection only, PPE.
340 Building Vault	RLW agitators	RLW,oil	--	X	X	X	X	X	X	Access through hole in cover block, covered by penthouse and locked. Pumps controlled from control room.	Entry for maintenance and inspection only, PPE.
340 Building Vault	RLW pumps	RLW	--	X	X	X	X	X	X	Access through hole in cover block, covered by penthouse and locked. Pumps controlled from control room.	Entry for maintenance and inspection only, PPE.
340 Building Vault, valve pit	RLW valves and piping	RLW	--	--	X	X	X	X	X	Access hatch, covered by penthouse and locked.	Entry for maintenance and inspection only, PPE.
340 Building Vault, valve pit	Sample/backup pump	RLW	--	X	X	X	X	X	X		
340 Building mechanical room	Air compressor, vacuum pump, MCC	Oil	X	X	--	X	--	--	--	Fire detection and suppression system.	Ear protection required.
340 Building decon room	Decon chemicals and equipment	Contamination, toxic chemicals	X	--	--	--	X	X	X	Steel grated trough drains to sump Fire detection and suppression system.	PPE as needed.
340 Building decon room	Sample hood	RLW, toxic chemicals	--	--	X	--	X	X	X	Ventilation.	PPE as needed.
340 Building extension	Solid waste packaging and storage	Solid mixed waste	X	X	--	X	--	--	X	Fire detection and suppression system.	--

Table 3-6. Operating Hazards. (2 sheets)

Room location	Equipment	Hazardous material	Hazards							Mitigating features	Administrative controls
			Fire	Electric shock	Air borne	Noise	Surface contamination	Skin contamination	Whole body dose		
340 Building control room	--	--	X	X	--	--	--	--	--	Fire detection and suppression system.	--
340-A Building	Storage tanks and valves	RLW	--	--	X	--	X	X	X	Bottom of tanks below ground.	Access tightly restricted, PPE.
340-B East Building	Waste loading	RLW	--	X	X	X	X	X	X	Shield wall, loading controlled from control room.	Access restricted, PPE during loading.
340-B West Building dry waste storage	drums, and boxes	Solid mixed waste	X	--	X	--	X	X	X	Shield wall separates solid waste storage area from tankcar loadout. Fire detection system.	--
MO-036 trailer	Facility Office	--	X	X	--	--	--	--	--	Fire Extinguishers.	--
MO-741 trailer	HPT office and survey station.	--	X	X	--	--	--	--	--	Fire Extinguishers.	--

HPT = health physics technician.
 MCC = motor control center.
 PPE = personal protective equipment.
 RLW = radioactive liquid waste.

Table 3-7. Maximum Individual Dose Due to an Aerosol Release of 1 Curie of Each Isotope From the 340 Facility. (2 sheets)

Receptor	Onsite 100 m north	Offsite 345 m East	Onsite 100 m north	Offsite 345 m East	
X/Q	4.21 E-02	3.77 E-03	4.21 E-02	3.77 E-3	
Isotope	Annual Effective Dose Equivalent		Maximum Organ		Organ
² H	1.30 E-03	1.16 E-04	1.60 E-03	1.43 E-04	Bone
¹⁴ C	2.90 E-02	2.60 E-03	3.70 E-02	3.31 E-03	R. Marrow
²² Na	1.20 E-01	1.07 E-02	1.80 E-01	1.61 E-02	Bone
⁵⁴ Mn	9.60 E-02	8.60 E-03	3.50 E-01	3.13 E-02	Lung
⁵⁷ Co	1.20 E-01	1.07 E-02	8.70 E-01	7.79 E-02	Lung
⁵⁹ Fe	1.80 E-01	1.61 E-02	7.30 E-01	6.54 E-02	Lung
⁶⁰ Co	2.80 E+00	2.51 E-01	1.80 E+01	1.61 E+00	Lung
⁶² Ni	3.20 E-02	2.87 E-03	1.60 E-01	1.43 E-02	Lung
⁶⁵ Zn	2.60 E-01	2.33 E-02	1.10 E+00	9.85 E-02	Lung
⁷⁵ Se	1.20 E-01	1.07 E-02	2.90 E-01	2.60 E-02	Lung
⁷⁶ Se	1.30 E-01	1.16 E-02	5.10 E-01	4.57 E-02	Lung
⁸⁶ Sr	2.60 E-02	2.33 E-03	5.10 E-02	4.57 E-03	Bone
⁹⁰ Sr	2.80 E+00	2.51 E-01	3.40 E+01	3.04 E+00	Bone
⁹⁰ Y	1.20 E-01	1.07 E-02	5.00 E-01	4.48 E-02	Lung
⁹³ Zr	1.10 E+00	9.85 E-02	2.80 E+01	2.51 E+00	Bone
⁹⁵ Zr	2.20 E-01	1.97 E-02	1.20 E+00	1.07 E-01	Bone
⁹⁵ Nb	8.70 E-02	7.79 E-03	4.80 E-01	3.84 E-02	Lung
⁹⁹ Tc	1.30 E-01	1.16 E-02	8.40 E-01	7.88 E-02	Lung
¹⁰⁶ Ru	6.60 E+00	5.91 E-01	5.30 E+01	4.75 E+00	Lung
¹⁰⁹ Cd	1.60 E+00	1.43 E-01	2.10 E+01	1.88 E+00	Kidney
¹²⁵ Sb	1.80 E-01	1.61 E-02	1.20 E+00	1.07 E-01	Lung
¹³¹ I	4.30 E-01	3.85 E-02	1.40 E+01	1.25 E+00	Thyroid
¹³⁴ Cs	6.10 E-01	5.46 E-02	7.30 E-01	6.54 E-02	Small int
¹³⁷ Cs	4.30 E-01	3.85 E-02	4.70 E-01	4.21 E-02	Small int
¹⁴⁴ Ce	5.20 E+00	4.66 E-01	4.10 E+01	3.67 E+00	Lung
¹⁵¹ Sm	4.20 E-01	3.76 E-02	7.10 E+00	6.36 E-01	Bone
¹⁵² Eu	2.90 E+00	2.60 E-01	1.80 E+01	1.61 E+00	Liver
¹⁵⁴ Eu	3.90 E+00	3.49 E-01	2.70 E+01	2.42 E+00	Bone
¹⁵⁵ Eu	5.70 E-01	5.10 E-02	7.60 E+00	6.81 E-01	Bone
¹⁸² Ta	6.20 E-01	5.55 E-02	4.30 E+00	3.85 E-01	Lung
²³⁴ U	1.90 E+03	1.70 E+02	1.50 E+04	1.34 E+03	Lung
²³⁵ U	1.70 E+03	1.52 E+02	1.40 E+04	1.25 E+03	Lung
²³⁸ U	1.80 E+03	1.61 E+02	1.50 E+04	1.34 E+03	Lung
²³⁸ U	1.70 E+03	1.52 E+02	1.40 E+04	1.25 E+03	Lung
²³⁹ Pu	4.00 E+03	3.58 E+02	3.80 E+04	3.40 E+03	Bone

Table 3-7. Maximum Individual Dose Due to an Aerosol Release of 1 Curie of Each Isotope From the 340 Facility. (2 sheets)

Receptor	Onsite 100 m north	Offsite 345 m East	Onsite 100 m north	Offsite 345 m East	
X/Q	4.21 E-02	3.77 E-03	4.21 E-02	3.77 E-3	
Isotope	Annual Effective Dose Equivalent		Maximum Organ		Organ
²³⁹ Pu	4.30 E+03	3.85 E+02	4.40 E+04	3.94 E+03	Bone
²⁴⁰ Pu	4.20 E+03	3.78 E+02	4.40 E+04	3.94 E+03	Bone
²⁴¹ Pu	6.90 E+01	6.18 E+00	9.30 E+02	8.33 E+01	Bone
²⁴² Pu	4.10 E+03	3.67 E+02	4.00 E+04	3.58 E+03	Bone
²⁴¹ Am	6.20 E+03	5.55 E+02	1.10 E+05	9.85 E+03	Bone
²⁴³ Cm	4.30 E+03	3.85 E+02	7.60 E+04	6.81 E+03	Bone
²⁴⁴ Cm	3.50 E+03	3.13 E+02	6.10 E+04	5.46 E+03	Bone

Table 3-8. Dose Consequences for a Spray-Release Accident.^(a)

Isotope ^(b)	Ci/L	Ci released	rem/Ci Onsite	rem/Ci Offsite	rem Onsite	rem Offsite
Beta/Gamma	6.90 E-01	3.53 E-01	3.90 E+00	3.30 E-01	1.38 E+00	1.16 E-01
Alpha	1.16 E-03	5.93 E-04	4.30 E+03	3.60 E+02	2.55 E+00	2.13 E-01
Total	--	--	--	--	3.92 E+00	3.30 E-01

(a) For a 5.11 E-4 L release while pumping from the 340 Vault tank to a railroad tankcar.

(b) For a worst-case release the Beta/Gamma distribution was modeled as ¹⁵⁴Eu and the Alpha was modeled as ²³⁹Pu.

Table 3-9. Dose Consequences When Burning the 340-B West Building Solid Wastes.

Isotope ^a	Total curies	Curies released	Onsite (rem/Ci)	Offsite (rem/Ci)	Onsite (rem)	Offsite (rem)
Beta/gamma	10.00 E+00	5.00 E-03	3.90 E+00	3.3 E-01	1.95 E-02	1.65 E-03
Alpha	1.50 E+00	7.50 E-04	4.30 E+03	3.60 E+02	3.23 E+00	2.70 E-01
Total					3.24 E+00	2.72 E-01

^a For a worst-case release the beta/gamma is taken as ¹⁵⁴Eu and the alpha is taken as ²³⁹Pu.

Table 3-10. 340 Facility Safety Enhancement Systems.

Component/system	Provisions	Codes, standards, and regulations
Stack Sampler/Monitoring System		
340 Building Stack	Continuous isokinetic sampler/no monitor.	ANSI N 13.1-1982
340 Vault Stack	Continuous isokinetic sampler. Continuous monitoring of beta particulates.	ANSI N 13.1-1982
Facility Confinement/Containment Systems		
340 Building Stack	Two-stage HEPA filters, one fan (no backup).	Topic Listing; Environmental Protection
340 Vault Stack	Two stages HEPA filters, demister, two fans.	
340-B Building Stack	One stage HEPA filters, one fan (no backup), only used during loadout activities.	
Fire Protection Systems		
340 Building	Smoke detectors, Temperature detectors, and sprinkler systems-alarms locally and at the Hanford Fire Department Central Dispatch Center.	DOE Order 5480.7A
340-B Building	Temperature activated fire detectors and sprinkler system alarms locally and at the Hanford Fire Department Central Dispatch Center.	

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4.0 CONFIGURATION CONTROLS SYSTEM

The configuration control system for the PHMC is administered using procedures, standards and practices under the Project Hanford Procedures listed in the Topic Listing, *Engineering Program*. The procedures in this Topic Listing govern all engineering, development and project tasks that result in a documented design and/or deliverable engineered product. These engineering practices provide a graded approach to the principles of configuration management and design control to ensure that engineering tasks and deliverable engineered products produced from these tasks are identified, authorized, verified, changed, and documented by a controlled methodology, that is consistent with the importance of the activities to safety, health, environment, continuity of operations, and value. Drawing controls, operating, and maintenance procedure controls, USQ, procedures and quality assurance procedures all provide assurance that functional and physical characteristics of the facility are maintained within the safety basis and that the ISB-003 is maintained current.

4.1 DRAWING CONTROL

Engineering drawings initiated and approved by or for the PHMC are prepared in accordance with HNF-PRO-1819, *PHMC Engineering Requirements*, and WHC-IP-1026, *Engineering Practices Guidelines*, (WHC 1994a). These procedures implement DOE Richland Operations Office (RL) Order 6430.1C, *Hanford Plant Standards*, Design Criteria, SDC-1.3, "Preparation and Control of Engineering and Fabrication Drawings." All released engineering drawings are stored in designated storage facilities. Before release, a designated reviewer ensures the drawing is in accordance with the above manuals. The approval process for drawings uses the engineering data transmittal form for recording approval signatures (*Engineering Data Transmittal Requirements*, HNF-PRO-244). Revision of approved drawings follow the procedures specified in *Engineering Document Change Control Requirements*, HNF-PRO-440. Approval of revised drawings use the Engineering Change Notice (ECN) form.

4.2 PROCEDURE CONTROL

The PHMC maintains a policy that all operations are to be conducted in accordance with written procedures. The requirement for complying with procedures is stated in HNF-PRO-033, *Employee Discipline*. Compliance with the detailed instructions in procedures is essential for safe and reliable operation. The procedures (maintenance and operating) will comply with the applicable requirements of DOE Orders 4330.4B and 5480.19, as stated in the requirements of HNF-PRO-224, *Document Control*, and HNF-PRO-229, *Technical Procedure Standard*.

These procedures and policies apply to all facilities and organizations that require the use of procedures for guidance and control of day-to-day activities, including the 340 Facility. Each facility is responsible for controlling, maintaining, and modifying the procedures to achieve the highest standards of safety, efficiency, and excellence in conduct of operations. The procedures are maintained current in procedure books that are readily available for the users. Outdated procedures are removed and replaced as

procedure revisions are approved and issued. Management accepts the responsibility of assuring that personnel using the procedures are trained and qualified.

4.3 UNREVIEWED SAFETY QUESTIONS

The USQ process complies with DOE Order 5480.21 and is administered by the PHMC using HNF-PRO-062, *Identifying and Resolving Unreviewed Safety Questions*. The USQ process for Waste Management Federal Services of Hanford, Inc. is established by procedure WMH-200, Section 4.3, "Unreviewed Safety Questions." The safety evaluation process makes use of the methodology and risk-acceptance guidelines contained in HNF-PRO-704, *Hazard and Accident Analysis Process*. This process allows the PHMC to make physical and procedural changes and conduct tests and experiments without prior DOE approval, as long as the changes do not affect the safety basis of the facility or result in a change to a technical safety requirement or IOSR. Conversely, the USQ evaluation process starts when changes to the operations and facility, a discovery of new conditions, or a discovery of an error in the safety analysis affects the safety basis of the facility or results in changes to a technical safety requirement or IOSR.

The USQ process also ensures that discoveries representing conditions outside the safety basis or technical safety requirements are identified. This is to ensure that actions are taken to modify the conditions or to redefine the safety basis.

The USQ process applies to activities at all PHMC-managed facilities (existing and new). The 340 Facility complies with the USQ process by evaluating proposed changes, tests, and experiments. If a USQ exists, a USQ safety evaluation will be performed and documented on the change. 340 Facility occurrences will be evaluated to determine if a discovery (potential USQ) might exist. If a discovery exists, the 340 Facility manager will be notified and actions will be taken to determine if the facility is in a safe (or safest) mode or state, based on the potential inadequacy or reduction in technical safety requirement margin. The appropriate RL Line manager will be notified when an occurrence is determined to be a discovery.

4.4 QUALITY ASSURANCE

The quality assurance program established by the PHMC for the design, construction, modification, and operation of the 340 Facility is consistent with the requirements of 10 CFR 830.120 *Quality Assurance*. The HNF-MP-599, *Project Hanford Quality Assurance Program Description* provides a basis for HNF-SP-1228, *Quality Assurance Program Implementation Plan for Nuclear Facilities* and HNF-SD-WM-QAPP-036, *Waste Management Federal Services of Hanford, Inc. Quality Assurance Program Plan*.

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5.0 REFERENCES

5.1 DOCUMENTS

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- Berneski, L.D., 1995, *Hanford 300 Area Retention Process Sewer - Inventory at Risk Calculations and Safety Analysis*, WHC-SD-WM-SAD-027, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- Blaak, T.M., 1998, *Waste Compatibility Assessment of 340 Facility Waste (340-98-01) With Tank 241-AP-106 Waste Via The 204-AR Facility*, Lockheed Martin Hanford Company, Richland, Washington.
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- HNF, 1998, *Building Emergency Plan, 340 Waste Management Facility*, HNF-IP-0263-340, Project Hanford Management Contract, Richland, Washington.
- WHC, 1991, *Health Physics Procedures Manual*, WHC-IP-0718, Westinghouse Hanford Company, Richland, Washington.
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- WHC, 1993, *Hanford Site Solid Waste Acceptance Criteria*, WHC-EP-0063, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1994a, *Engineering Practices Guidelines*, WHC-IP-1026, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1994b, *300 Area Effluent Facilities Administrative Manual*, WHC-IP-1000, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1996a, *340 Facility Waste Tank System Integrity Assessment Report*, WHC-SD-WM-ER-546, Rev. 0, E.J. Walter, Westinghouse Hanford Company, Richland, Washington.
- UBC, 1991, *Uniform Building Code*, International Conference of Building Officials, Whittier, California.

5.2 U.S. FEDERAL CODES AND DEPARTMENT OF ENERGY ORDERS/GUIDELINES

- 10 CFR 830.120, *Quality Assurance*, Code of Federal Regulations, as amended.
- 10 CFR 835, *Occupational Radiation Protection*, Code of Federal Regulations, as amended.
- DOE/EA-1178, 1997, *Environmental Assessment, 300 Area Steam Plant Replacement, Hanford Site, Richland, Washington*, U.S. Department of Energy, Washington, D.C.
- DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, U.S. Department of Energy, Washington, D.C.
- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, U. S. Department of Energy, Washington, D.C.
- DOE-STD-3011-94, *Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans*, U. S. Department of Energy, Washington, D.C.
- DOE Order 5480.21, *Unreviewed Safety Questions*, U.S. Department of Energy, Washington, D.C.
- DOE Order 5480.23, *Nuclear Safety Analysis Reports*, U.S. Department of Energy, Washington, D.C.
- DOE Order, 5480.24, *Nuclear Criticality Safety*, U.S. Department of Energy, Washington, D.C.
- DOE Order, 5481.1B, *Safety Analysis and Review System*, U.S. Department of Energy, Washington, D.C.
- DOE Order, 5482.1B, *Environmental, Safety, and Health Appraisal Program*, U.S. Department of Energy, Washington, D.C.

DOE/RL-96-109, Rev. 0, *Hanford Radiological Control Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

5.3 WESTINGHOUSE HANFORD COMPANY CONTROLLED MANUALS

The WHC controlled manuals have all been superseded by Project Hanford Procedures. The references are retained in this document to provide a historical perspective.

HSRCM-1, *Hanford Site Radiological Control Manual*, Hanford Contractors Richland, Washington.

WHC-CM-1, *Company Policies and Charters*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-1-3, *Management Requirements and Procedures*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-1-10, *Safety Manual*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-1-11, *Industrial Hygiene Manual*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-3-5, *Document Control and Records Management*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-1, *Fire Protection Program*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-2, *Quality Assurance*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-29, *Nuclear Criticality Safety*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-33, *Security*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-41, *Fire Protection Program*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-43, *Emergency Management Procedures*, Westinghouse Hanford Company, Richland, Washington.

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

WHC-CM-6-1, *Standard Engineering Practices*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-7-5, *Environmental Compliance*, Westinghouse Hanford Company, Richland, Washington.

5.4 PROJECT HANFORD PROCEDURES--This list is not intended to be a comprehensive listing of all Project Hanford procedures that may apply to the 340 Facility.

Topic Listings

- *Emergency Management*
- *Engineering Program*
- *Environmental Protection*
- *Fire Protection*
- *Nuclear and System Safety*
- *Occupational Safety & Health*
- *Quality Assurance*
- *Safeguards & Security*
- *Transportation and Packaging*
- *Waste Management*

Project Hanford Procedures

HNF-PRO-033,	<i>Employee Discipline</i>
HNF-PRO-062,	<i>Identifying and Resolving Unreviewed Safety Questions</i>
HNF-PRO-069,	<i>Maintenance Management</i>
HNF-PRO-072,	<i>Plant Instrumentation and Equipment Functional Status Identification</i>
HNF-PRO-082,	<i>Radiological Worker Training</i>
HNF-PRO-224,	<i>Document Control</i>
HNF-PRO-229,	<i>Technical Procedure Standard</i>
HNF-PRO-242,	<i>Engineering Drawing Requirements</i>
HNF-PRO-244,	<i>Engineering Data Transmittal Requirements</i>
HNF-PRO-334,	<i>Criticality Safety General Requirements</i>
HNF-PRO-430,	<i>Safety Analysis Program</i>
HNF-PRO-440,	<i>Engineering Document Change Control Requirements</i>
HNF-PRO-546,	<i>Criticality Alarm System</i>
HNF-PRO-549,	<i>Criticality Safety Nonconformance Response</i>
HNF-PRO-700,	<i>Safety Analysis and Technical Safety Requirements</i>
HNF-PRO-701,	<i>Safety Analysis Process-Existing Facility</i>
HNF-PRO-704,	<i>Hazard and Accident Analysis Process</i>
HNF-PRO-1619,	<i>ALARA Organization and Responsibilities</i>
HNF-PRO-1819,	<i>PHMC Engineering Requirements</i>
HNF-MP-599,	<i>Project Hanford Quality Assurance Program Description</i>
HNF-SP-1228,	<i>Quality Assurance Program Implementation Plan for Nuclear Facilities</i>

5.5 WASTE MANAGEMENT FEDERAL SERVICES OF HANFORD PROCEDURES--This list is not intended to be a comprehensive listing of all WMH procedures that may apply to the 340 Facility.

HNF-SD-WM-QAPP-036, *Waste Management Federal Services of Hanford, Inc. Quality Assurance Program Plan*

WMH-100, *Waste Management Hanford Policies*

WMH-200, *Waste Management Hanford Procedures*

WMH-200, Section 3.1, "Waste Management Hanford Procedure"

WMH-200, Section 3.2, "ALARA Program"

WMH-200, Section 3.8, "Enhances Radiological Work Planning"
WMH-200, Section 4.3, "Unreviewed Safety Questions"

HNF-IP-1000, *300 Area Liquid Effluent Facilities Administration*
HNF-IP-1000, Section 2.1, "Operations Organization and Administration"
HNF-IP-1000, Section 2.2, "340 Facility Equipment/System Deactivation Lay-UP
Status and Control"

APPENDIX A

340 FACILITY HYDROGEN GENERATION ESTIMATES

FLUOR DANIEL NORTHWEST, INC.

1100 Jadwin
P.O. Box 1050
Richland, Washington 99352-1050

June 9, 1998

Waste Management Projects
KRH-SLB-A31630

CO-98-113

Mr. R. M. Hadley, Manager
Contracts and Procurement
Waste Management Federal Services of Hanford, Inc.
P. O. Box 700
MSIN G1-58
Richland, Washington 99352-0700

Dear Mr. Hadley:

**TASK ORDER D06 - TRANSMITTAL OF DELIVERABLE, 340 FACILITY HYDROGEN
GENERATION ESTIMATES**

Response Due: N/A

Responds To: Letter, R. M. Hadley, Waste Management Federal Services of Hanford, Inc., to E. R. Jacobs, Fluor Daniel Northwest, Inc., "Contract KRH-SLB-D31630 Request for Task Order 06 Proposal - Staff Augmentation, Nuclear Safety Engineer and Criticality Safety Specialist," WMH-9853912, dated May 4, 1998 (CI-98-081)

Fluor Daniel Northwest, Inc. (FDNW) is pleased to attach the following deliverables:

340 Facility Hydrogen Generation Estimates

Unless otherwise directed, FDNW will allow this Task Order to remain open. This will allow Waste Management Federal Services of Hanford, Inc. the opportunity to perform additional work, if desired.

FDNW appreciated the opportunity to support WMH on this effort. Please contact Rex Bendixsen at 376-9523 or Robert Marusich at 376-8081 if you have any questions.

Sincerely,

D. R. Lucas
Area Manager

DRL:rst

Attachment

340 FACILITY HYDROGEN GENERATION ESTIMATES

Prepared by: Rex B. Bendixsen Date: 6/9/98

Rex B. Bendixsen
SAR Engineering Services
Fluor Daniel Northwest

And: Robert M. Marusich Date: 6/9/98

Robert M. Marusich
Safety Analysis and Risk Assessment
Fluor Daniel Northwest

Approved by: Larry E. Johnson Date: 6/9/98

Larry E. Johnson, Manager
SAR Engineering Services
Fluor Daniel Northwest

340 FACILITY HYDROGEN GENERATION ESTIMATES

I INTRODUCTION

The 340 Facility, located in the 300 Area, provides temporary storage for radioactive liquid waste generated by several buildings. The radioactive liquids generated by the Pacific Northwest National Laboratory buildings gravity flow into the 340 Facility Vault Tanks. Radiolysis of liquids generates hydrogen gas which rises to the liquid surface. The hydrogen is diluted to safe concentrations by process ventilation air flows and is exhausted through the 340 Facility ventilation stack. The 340 Facility requested estimates of hydrogen generation rates and equilibrium levels for the vault storage tanks using both the Interim Safety Basis source term and the radionuclide concentrations currently being collected in the 340 Facility Vault Tanks. This report documents the hydrogen generation estimates provided by R. B. Bendixsen and R. M. Marusich of Fluor Daniel Northwest, Inc., in June, 1998.

II SUMMARY

Estimates of process air flow rates and hydrogen generation rates using the vault tank bounding source term demonstrate that the hydrogen in air concentrations of not greater than 0.01 volume percent is much less than the lower hydrogen flammability limit - 4 volume percent. In addition, the ventilation system backup exhaust fan and absence of ignition sources provide further assurance that hydrogen deflagration has a low potential for occurrence. Also, the current curie content of waste received in the vault tanks is much less than the bounding source term. Therefore, adequate controls assuring safe hydrogen in air concentrations are maintained and additional controls are not needed. The attached Design Calculation sheets contain the assumptions and calculations.

III RECOMMENDATIONS

1. Update the 340 Facility Interim Safety Basis Document, WHC-SD-WM-ISB-003, Rev. 1-D, by adding the following paragraph to Section 3.1.3.7.

"Radiolysis of liquid in the vault tanks generates hydrogen gas which rises to the liquid surface, is diluted to safe concentrations by the process vent flow, and exits the facility through the ventilation stack. Estimates of process air flow rates and hydrogen generation rates using the tank bounding source term (Table 2-3) demonstrate that the hydrogen in air concentration of no greater than .01 volume percent is much less than the lower hydrogen flammability limit - 4 volume percent (Letter, *340 Facility Hydrogen Generation Estimates*, 1998, Appendix E). In addition, the ventilation system backup exhaust fan and absence of ignition sources provide additional assurance that hydrogen deflagration has a low potential for occurrence. Also, the current curie content of waste received in the vault tanks is much less than the bounding source term (Appendix E). The hazards for this potential accident are summarized in Table 3-4, V.I.D.

Assuming loss of power to the exhaust fans, the hydrogen in air concentrations would reach 50% of the lower explosive limit in about 10 days (Appendix E) using the Table 2-3 source term. This gives operators enough time to provide emergency power to the exhaust fans and maintain safe hydrogen concentrations. Therefore, adequate controls assuring safe hydrogen in air concentrations are maintained and additional controls are not needed."

2. Add this letter as an appendix to the ISB.
3. Add the attached table to Table 3-4 of the ISB, "VI Radiation"

Table 3-4. 340 Waste Handling Facility Preliminary Hazard Analysis. (6 sheets)

Hazard/ energy source	Potential accident and sequences	Target/potential consequences	Mitigating features		Annual probability	Severity
			Engineered	Administrative		
D. Hydrogen	Radiolysis of water generates hydrogen, the hydrogen accumulates and is ignited	Property damage, personnel injury, and radiation exposure could occur	Tank ventilation air flow dilutes hydrogen below flammable concentrations Backup exhaust blower	Procedures provide for recovery actions on loss of vent flow	Unlikely	III
			Hydrogen generation rate is low No ignition source in tank or process vent system			

IV REFERENCES

- Hu, A. T., 1997, *Calculations of Hydrogen Release Rate at Steady State Conditions for Double Shell Tanks*, HNF-SD-WM-CN-117, Lockheed Martin Company, Richland, Washington.
- PB-230 846, *Radiological Health Handbook*, 1970, U.S. Department of Health, Education, and Welfare, Rockville, Maryland.
- RHO-WM-EV-9P, *Hydrogen Control in the Handling, Shipping and Storage of Wet Radioactive Waste*, 1985, Rockwell Hanford Operations, Richland, Washington.
- WHC-SD-WM-ISB-003 Rev. 1-D, *340 Waste Handling Facility Interim Safety Basis*, 1998, Waste Management Hanford, Richland, Washington.

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 1 of 10
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject 340 Facility H₂ Generation and Tank Concentration
 (8) Originator R M Marovich Date _____
 (9) Checker _____ Date _____

(10)

The 340 Facility contains a Waste Neutralization Tank. The tank typically contains 21,000 L of waste but can and presently does contain 55,000 L. The 55000 L value is tank capacity.

The tank dimensions are as follows

14 ft ID

13'-1" tall

Flat top, cylindrical tank with a nearly flat bottom.

Assume the bottom is flat.

The tank volume is found from

$$\begin{aligned}
 V &= \frac{\pi}{4} (D)^2 (H) = \frac{\pi}{4} (14 \text{ ft})^2 (13.08 \text{ ft}) \\
 &= 2013.5 \text{ ft}^3 \\
 &= (2013.5 \text{ ft}^3) \left(\frac{28.3 \text{ L}}{\text{ft}^3} \right) \\
 &= 56982 \text{ L}
 \end{aligned}$$

call it 57,000 L.

Hydrogen generation requires knowledge of radionuclide concentration and chemical concentration.

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 2 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) Two cases will be considered.

a) ISB Case

b) "Average" Case

The ISB case comes from the 340 Facility ISB (WHC-SD-WM-ISB-003, Rev 1 author R.T. Stordewar 10/96, Westinghouse Hanford Company), Table 2-3

Table 2-3 states the following

Bounding Source Term

β/γ 0.69 Ci/L

α 1.16×10^{-3} Ci/L

Table 2-1 of the ISB shows that of the β/γ emitters the radionuclides having the greatest concentrations are:

^{137}Cs (85% of total) $2.37\text{E}-2$ Ci/L

^{154}Eu (28% of total) $7.9\text{E}-3$ Ci/L

^{152}Eu $4.49\text{E}-3$ Ci/L

^{241}Pu $7.48\text{E}-3$ Ci/L

The α emitters are dominated by:

^{238}Pu $2.72\text{E}-4$ Ci/L

^{241}Am $6.26\text{E}-4$ Ci/L

^{240}Pu $7.85\text{E}-4$ Ci/L

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 3 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

- (10) Table 2-1 shows data from rail shipping records. It is assumed that the radionuclide distribution will be the same for the ~~max~~ bounding case.

To determine hydrogen generation, the energy deposition is needed. This means that the γ and β energies must be known. These are multiplied by C/L and by L of waste to get total energy deposited. The generation rate is a function of energy deposited from the Radiological Health Handbook, 1970 Edition

137 Cs	0.514 meV β (100%)	0.662 meV γ (85%)
	1.176 meV (1.7%)	
	0.624 meV e^- (100 ¹⁰ %)	
154 Eu	1.85 meV β (10%)	1.278 meV γ (37%)
	0.87 meV (100%)	1.00 meV (81%, doublet)
		0.576 meV (12%)
152 Eu	1.48 meV β (100%)	1.408 meV (12%)
	0.120 meV e^- (200%)	1.113 meV (14%)
		1.087 meV (12%)
		0.965 meV (15%)
		0.779 meV (14%)
		0.344 meV (27%)
²⁴¹ Pu	0.021 meV β	

energies are maximum energies. Average $\approx$ max / 3
 e^- energies are single values

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 4 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

- (10) Since the radionuclide concentration is dominated by a ^{137}Cs , it will be assumed that the average γ energy ~~and~~ and β energy is:

γ 0.7 meV (100%) to account for the $>1\text{ meV } \gamma$ from Eu.

β 1.25 meV (100%)
 (e⁻ actually)

For ease of calculation I will use 1 MeV for the γ , β energy and ignore any γ losses from the tank. (See Appendix 1 as well for further justification of use of 1 MeV).

For the α case use 5.5 meV.

ISB case - total energy deposited
 55,000 L case

$$E = \left(0.69 \frac{\text{Ci}}{\text{L}}\right) \left(3.7 \times 10^{10} \frac{\text{dis}}{\text{s} \cdot \text{Ci}}\right) \left(55,000 \text{ L}\right) \left(\frac{1.06 \text{ eV}}{\text{dis}}\right) +$$

$$\left(1.16 \times 10^{-3} \frac{\text{Ci}}{\text{L}}\right) \left(3.7 \times 10^{10} \frac{\text{dis}}{\text{s} \cdot \text{Ci}}\right) \left(55,000 \text{ L}\right) \left(\frac{5.5 \times 10^6 \text{ eV}}{\text{dis}}\right)$$

$$= 1.4 \times 10^{21} \frac{\text{eV}}{\text{s}} + 1.8 \times 10^{19} \frac{\text{eV}}{\text{s}} = 1.42 \times 10^{21} \frac{\text{eV}}{\text{s}}$$

ISB, 55,000 L case, $1.4 \times 10^{21} \text{ eV/s}$

to 2 significant figures

Similarly the 21,000 L case

ISB, 21,000 L case, $5.4 \times 10^{20} \text{ eV/s}$

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 5 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) Average Case

The ~~radio~~ radiation radionuclide concentration under average conditions comes from a recent spread sheet from the plant.

$\beta\gamma$ The dominant radionuclides are

$$^{137}\text{Cs} \dots 1.14 \text{E-1 } \mu\text{Ci/ml}$$

$$^{90}\text{Sr} \dots 3.09 \text{E-2 } \mu\text{Ci/ml}$$

$$^{241}\text{Pu} \dots 2.44 \text{E-2 } \mu\text{Ci/ml}$$

$$\text{total} = 1.84 \text{E-1 } \mu\text{Ci/ml}$$

$$= 1.8 \times 10^{-4} \text{ Ci/L}$$

$$\alpha \text{ total} = 2.88 \times 10^{-3} \mu\text{Ci/ml}$$

$$= 2.9 \times 10^{-6} \text{ Ci/L}$$

Since the α concentration is about 1/100 lower and energy is only 5% greater, The energy deposition will be a factor of

0.69 / 1.8E-4 or 3833 less, as just the $\beta\gamma$ portion are important, using the same energies and equations as above, it is seen that:

$$\text{Average, } 55,000 \text{ L, } 3.65 \times 10^{17} \text{ ev/s}$$

$$\text{Average, } 21,000 \text{ L, } 1.4 \times 10^{17} \text{ ev/s}$$

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 6 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

- (10) The generation rate for hydrogen is a function of the chemical concentrations of nitrate and nitrite per Henrie JO, "Hydrogen Control in the Handling, Shipping and Storage of Wet Radioactive Waste", RHO-WM-EU-9P, 1985 Rockwell Hanford Co, Richland WA. Organics are also important.

The ISB chemical concentration comes from Table 2-2.

Table 2-2 shows

$$\text{NO}_3^- = 3.4 \text{ E}+4 \text{ mg/L} = 0.55 \text{ moles/L}$$

$$\text{NO}_2^- = 3.1 \text{ E}+3 \text{ mg/L} = 0.07 \text{ moles/L}$$

$$\text{TOC} = 3.6 \text{ g/L}$$

The hydrogen generation rate is given by

$$G(\text{H}_2) = 0.45 - 0.31 [\text{NO}_2^-]^{1/3} - 0.41 [\text{NO}_3^-]^{1/3} + 0.15 [\text{TOC}]$$

$$\text{where } [\text{NO}_2^-] [\text{NO}_3^-] = \text{moles/L}$$

$$[\text{TOC}] = \text{wt\% in solution}$$

$$G(\text{H}_2) = \frac{\text{molecules of H}_2}{100 \text{ ev}}$$

Hu, AT, 1997, Calculations of Hydrogen Release Rate at Steady State Conditions for Double Shell Tanks", HNF-SD-WM-CW-117, Lockheed Martin.

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 7 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) He also states that in the term

$$0.45 - 0.31 [\text{NO}_2]^{1/3} - 0.41 [\text{NO}_3]^{1/3}$$

is less than 0.031, it is set equal to 0.031

$$0.45 - 0.31 [0.07]^{1/3} - 0.41 [\text{NO}_3]^{1/3} =$$

$$0.45 - 0.13 - 0.33 < 0.031$$

$$\text{So } G(\text{H}_2) = 0.031 + 0.15 \left(\frac{3.6}{1000} \right)^{100}$$

$$= 0.085$$

The density of this solution is found as follows

$$\text{NO}_3 \quad \frac{34 \text{ g}}{\text{L}} = \frac{46.8 \text{ g NaNO}_3}{\text{L}}$$

$$= \frac{20.7 \text{ cc}}{\text{L}} \quad \text{using } \rho = 2.26 \text{ g/cc}$$

$$\text{NO}_2 \quad \frac{3.1 \text{ g}}{\text{L}} = \frac{4.8 \text{ g NaNO}_2}{\text{L}}$$

$$= \frac{2.2 \text{ cc}}{\text{L}} \quad \text{using } \rho = 2.2 \text{ g/cc}$$

Water makes up

$$1000 \text{ cc} - 2.2 - 20.7 = 977 \text{ cc}$$

$$\text{wt of solution} = \frac{(977 \text{ cc}) \left(\frac{1 \text{ g}}{\text{cc}} \right) + (46.8 \text{ g}) + (4.8 \text{ g})}{1000 \text{ cc}} = \frac{1.03 \text{ g}}{\text{cc}}$$

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 8 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) Using Figure 4-5 of Henrie 1985

$$\text{get } G(\text{CH}_2) = \frac{0.14 \text{ molecules}}{100 \text{ eV}}$$

then adding in the TOL component yield

$$\text{ISB } G(\text{CH}_2) = \frac{0.085 \text{ molecules}}{100 \text{ eV}}$$

This is the value that will be used

For the Average case

$$[\text{NO}_3] = 9.4 \text{E}+3 \text{ ppm}$$

$$= \frac{9.4 \text{E}+3 \text{g}}{10^6 \text{g}} = \frac{9400 \text{g}}{1000 \text{L}} = \frac{9.4 \text{g}}{\text{L}} = 0.15 \text{ moles/L}$$

$$[\text{NO}_2] = 279 \text{ ppm}$$

$$= \frac{279 \text{g}}{10^6 \text{g}} = \frac{279 \text{g}}{1000 \text{L}} = \frac{0.28 \text{g}}{\text{L}} = 0.006 \text{ moles/L}$$

$$[\text{TOL}] = 148 \text{ ppm}$$

$$= \frac{0.148 \text{g}}{\text{L}}$$

water volume
density =

$$\left(\frac{0.15 \text{ moles}}{\text{L}} \right) \left(\frac{85 \text{g}}{\text{mole}} \right) \left(\frac{1 \text{L}}{2.26 \text{g}} \right) + \left(\frac{0.006 \text{ moles}}{\text{L}} \right) \left(\frac{69 \text{g}}{\text{mole}} \right) \left(\frac{1 \text{L}}{2.2 \text{g}} \right) = 5.8 \text{cc}$$

water moles up 994cc

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 9 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) density is then

$$\frac{994 \text{ g} + \left(\frac{0.15 \text{ moles}}{1} \right) \left(\frac{85 \text{ g}}{\text{mole}} \right) + \left(\frac{0.006 \text{ moles}}{1} \right) \left(\frac{6.9 \text{ g}}{\text{mole}} \right)}{1010 \text{ cc}} = \frac{1.007 \text{ g}}{\text{cc}}$$

call it $\frac{1 \text{ g}}{\text{cc}}$

From Henrie

$$G(\text{H}_2) = \frac{0.45 \text{ molecules}}{100 \text{ eV}}$$

adding in the T.O.C

$$G(\text{H}_2) = 0.45 + 0.15(0.0148) \\ = \cancel{0.45} 0.45 \text{ molecules} \\ 100 \text{ eV}$$

The $G(\text{H}_2)$ for α is about double that for β, δ except the energy deposition for α is very much smaller. Therefore the α contribution is negligible.

HNF-SD-WM-ISC-003, REV 2
DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 10 of 16
(4) Building _____ (5) Rev. _____ (6) Job No. _____
(7) Subject _____
(8) Originator _____ Date _____
(9) Checker _____ Date _____

(10)

Cases

Document	Waste Volume, L	G(CH ₄)	energy deposited	H ₂ generation rate L/min
ISB	55,000	0.085 ^{molecules} _{100ev}	1.4×10^{21} ev/s	2.7×10^{-3}
	21,000	0.085	5.4×10^{20} ev/s	1.0×10^{-3}
Average	55,000	0.45	3.65×10^{17} ev/s	3.7×10^{-6}
	21,000	0.45	1.4×10^{17} ev/s	1.4×10^{-6}

① H₂ generation rate

$$\left(\frac{\text{energy deposited}}{\text{ev/s}} \right) \left(\frac{G(\text{CH}_4)}{100 \text{ ev}} \right) \left(\frac{\text{mole}}{6.02 \times 10^{23} \text{ molecules}} \right) \left(\frac{22.4 \text{ L}}{\text{mole}} \right) \left(\frac{60 \text{ s}}{\text{min}} \right)$$

head space concentration

$$\frac{dm_H}{dt} = \dot{m}_{H,in} - \dot{m}_{H,out}$$

m_H = mass of hydrogen in head space, g

$\dot{m}_{H,in}$ = mass flow rate of hydrogen in, g/min
 $= \left(\frac{\text{L}}{\text{min}} \right) \left(\frac{0.09 \text{ g}}{\text{L}} \right)$

~~mass~~ $\dot{m}_{H,out}$ = mass flow rate out

= (volumetric flow rate in + vent flow rate out) $\left(m_H \right) \left(\frac{1}{V} \right)$

V = head space volume, L

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 11 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) At steady state

$$\left(\frac{L}{\text{min}}\right)_{\text{in}} \left(\frac{0.09g}{L}\right) = \left(\frac{L}{\text{min}}\right)_{\text{in}} + \frac{L}{\text{min}}_{\text{vent}} \left(\frac{m_H}{V}\right)$$

~~Assume~~ Assume vent flow rate of $1 \text{ ft}^3/\text{min} = \frac{28.3L}{\text{min}}$
 For the FSB, 58,000 L case

$$V = 2000 \text{ L} \quad (\text{from } 57,000 \text{ L} - 55000 \text{ L})$$

$$\frac{L}{\text{min}}_{\text{in}} = 2.7 \times 10^{-3}$$

$$m_H = 0.017g$$

$$\text{Vol H} = \frac{0.017g}{0.09g/L} = 0.19L$$

$$\text{Vol \%} = \frac{0.19L}{2000L} = 0.01\%$$

If 100 ft³/min were used (2830 L/min) Vol % = 10.4 % H₂

• Assume atmospheric breathing

$$0.4 \text{ vol \% / day} =$$

$$\frac{0.4}{100} \left(\frac{2000L}{\text{day}}\right) \left(\frac{\text{day}}{24 \text{ hr}}\right) \left(\frac{\text{hr}}{60 \text{ min}}\right) = 5.56 \times 10^{-3} \frac{L}{\text{min}}$$

Now solve for m_H

$$m_H = 58.9g$$

$$\text{Vol H} = \frac{58.9g}{0.09g/L} = 654L$$

$$\text{Vol \%} = \frac{654}{2000} = 33\%$$

Actual
Flow rate
is 150 cfm
per pg 15.

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 12 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) H_2 vol % (atmospheric breathing)

Cave	$\cancel{V_{in}}$ L/min	V_{out} breathing, L/min	Headspace Volume, L	Vol % H_2	
ISB	55,000 L	2.7×10^{-3}	5.56×10^3	2000	33%
	21,000 L	1.0×10^{-3}	0.1	36,000	19%
Average	55000 L	3.7×10^{-6}	5.56×10^3	2000	0.07%
	21,000 L	1.4×10^{-6}	0.1	36,000	$1.4 \times 10^{-3} \%$

For ISB condition 19% H_2 occurs as a steady state Headspace concentration for 21,000 L of waste.

Can get 19% H_2 in 2000 L of $V_{in} = ?$

19% $H_2 = 25\%$ of the LFL

$$(2000 L)(0.01) = 20 L$$

$$= 1.8 g-H_2$$

$$\left(\frac{L}{min} \right)_{in} \left(\frac{0.09 g}{L} \right) = \left(\frac{L}{min} \right)_{in} + 5.56 \times 10^3 \left(\frac{1.8 g}{2000 L} \right)$$

$$\frac{L}{min} \Big|_{in} = 5.6 \times 10^{-5} \frac{L}{min}$$

or 48 times less.

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 13 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10)

So if the ISB source term were a factor of 48 less or

$$0.014 \frac{\text{Ci}}{\text{L}} \quad \beta: \gamma$$

$$2 \times 10^{-5} \frac{\text{Ci}}{\text{L}} \quad \alpha$$

steady state H_2 concentration of 1% by volume would occur in 5500 L of waste, tank breathing condition

How long can vent system be out

$$\frac{dm_H}{dt} = \dot{m}_{H,in} - \frac{\dot{V}}{V} m_H$$

$$m_H(t) = \frac{\dot{m}_{H,in}}{\dot{V}/V} (1 - e^{-(\dot{V}/V)t})$$

$$\dot{m}_{H,in} = 2.7 \times 10^{-3} \frac{\text{L}}{\text{min}} (0.09 \frac{\text{g}}{\text{L}}) = 2.4 \times 10^{-4} \text{ g/min}$$

$$\dot{V} = 5.56 \times 10^{-3} \text{ L/min} + 2.7 \times 10^{-3} \text{ L/min} = 8.26 \times 10^{-3} \text{ L/min}$$

the "L/min in" term is ^{NIT} small in comparison

$$V = 2000 \text{ L}$$

$$m_H \text{ at } 1\% \text{ H}_2 = 1.8 \text{ g}$$

$$t = \frac{1.8 \text{ g}}{2.4 \times 10^{-4} \text{ g/min}} = 7500 \text{ min}$$

5 d outage max
to get to 1% H_2

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 14 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

(10) $1\% \text{ H}_2$ is 25% of the LPL

50% of the LPL = 2% H_2 in head space

$$(0.02)(2000 \text{ L}) = 40 \text{ L}$$

$$40 \text{ L} \left(\frac{0.09 \text{ g}}{\text{L}} \right) = 3.6 \text{ g}$$

$$t = 15485 \text{ min} = 258 \text{ hr} \\ = 10.8 \text{ days}$$

Vent can be out for 5 days before
 25% of the LPL is reached and
 10 days before 50% of the LPL is
 reached assuming

55,000 L waste

ISB source term

0.6 M NO_3 , NO_2

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 15 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____

Estimated Flow Rate For 340 Facility Vault Tanks (2)

Data: Heater Current 1.9 Amps

Current off - downstream thermocouple 60.9 to 61.4 °F

Current on - downstream thermocouple 89.9 to 91.0 °F

Voltage 490-495 volts

Heater Rating 2 kW 'Y' connected

Three phase Circuit Power $W = V_{\text{line}} I_{\text{line}} \sqrt{3} \cos \theta$

Assume Power Factor (Efficiency) $\cos \theta = .9$

$W = 490(1.9)\sqrt{3}(.9) = 1,450 \text{ Watts}$ Compare 1.45 kW = 2.0 Heater Rating

Air Flow Temperature Rise

$$H = W C_p \Delta T$$

$$\text{Solve for } W \quad W = \frac{H}{C_p \Delta T}$$

$$W = \frac{1,450 \text{ watt} \times \frac{Btu/hr}{1,293 \text{ W}} \times \frac{1 \text{ hr}}{60 \text{ min}} \times \frac{1}{29 \text{ ft}} \times \frac{359 \text{ ft}^3}{\text{lbmole}}}{.24 \frac{Btu}{\text{lbm} \cdot ^\circ \text{F}} \times (89.9 - 61.4)^\circ \text{F}} = 149 \frac{\text{ft}^3}{\text{min}}$$

$$\therefore W = 149 \frac{\text{ft}^3}{\text{min}}$$

$$\text{I.e. } V = 495 \text{ volts } W = 149 \times \frac{495}{990} = 150 \frac{\text{ft}^3}{\text{min}} \quad \left. \begin{array}{l} \\ \end{array} \right\} \text{Standard ft}^3/\text{min}$$

$$\frac{1}{18} \Delta T = (91 - 60.9) = 30.1^\circ \text{F} \quad W = \frac{149 \times 30.1}{28.5} = 157 \frac{\text{ft}^3}{\text{min}}$$

$$(P) \quad 60.9^\circ \text{F} \text{ H. Actual } \frac{\text{ft}^3}{\text{min}} = 149 \times \frac{(91.0 + 60.9)}{460} = 169 \frac{\text{ft}^3}{\text{min}} \quad \text{Compare this to plant heat value (500 CFM)}$$

BB²
5/19/98

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 16 of 16
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator _____ Date _____
 (9) Checker _____ Date _____

Appendix 1

(10) Detailed energy deposition

137

Cs

$$(0.514 \text{ mev } \beta/\gamma) = 0.171$$

$$(1.176 \text{ mev } \beta/\gamma)(0.07) = 0.027$$

$$(0.624 \text{ mev } e^-)(0.1) = 0.0624$$

$$(0.662 \text{ mev } \gamma)(0.85) = 0.563$$

$$0.824 \text{ mev/dis}$$

154 Eu

$$(1.85 \text{ mev } \beta/\gamma)(0.1) = 0.062$$

$$(0.87 \text{ mev } \beta/\gamma)(1.0) = 0.29$$

$$(1.28 \text{ mev } \gamma)(0.37) = 0.473$$

$$(1 \text{ mev } \gamma)(2)(0.31) = 0.62$$

$$(0.876 \text{ mev } \gamma)(0.12) = 0.105$$

$$1.55 \text{ mev/dis}$$

152 Eu

$$(1.48 \text{ mev } \beta/\gamma)(1.0) = 0.49$$

$$(0.12 \text{ mev } e^-)(2.00) = 0.24$$

$$(1.408 \text{ mev } \gamma)(0.12) = 0.17$$

$$(1.113 \text{ mev } \gamma)(0.14) = 0.156$$

$$(1.087 \text{ mev } \gamma)(0.12) = 0.13$$

$$(0.965 \text{ mev } \gamma)(0.15) = 0.145$$

$$(0.779 \text{ mev } \gamma)(0.14) = 0.11$$

$$(0.344 \text{ mev } \gamma)(0.22) = 0.09$$

$$1.54 \text{ mev/dis}$$

241 Pa

$$(0.021 \text{ mev } \beta/\gamma) = 0.07$$

$$\frac{(0.824)(2.37E-2) + 1.55(7.9E-3) + 1.54(4.49E-3) + 0.07(7.48E-3)}{2.37E-2 + 7.9E-3 + 4.49E-3 + 7.48E-3} = 0.9 \text{ mev/dis on the ISB average}$$

FLUOR DANIEL NORTHWEST

SENIOR ANALYSTS ADVISORY GROUP

CHECKLIST FOR PEER REVIEW

Document Reviewed: 340 Facility H₂ Generation And Tank Concentration, Design Calculation Sheets (16).

Scope of Review: ENTIRE DOCUMENT

Yes No NA

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

RB BENDTSEN *RB Bendtsen*

6/8/98

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.

APPENDIX B
CALCULATION OF SPRAY RELEASE

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CALCULATION OF SPRAY RELEASE

Calculation of the spray release from the Vault Tank piping used equations from WHC-SD-GN-TI-30003 and WHC-SD-GN-SWD-20007. The calculations were done on a spreadsheet iterating first to find the Reynolds number and then to find the slit width giving the highest respirable flow. The Reynolds number 3230 indicates a critical flow. Subscripts were changed in some of the equations for consistency.

The following variable constants define the system.

Velocity Coefficient	$C_v =$	8.20 E-01	-	
Resistance Coefficient	$K_v =$	4.87 E-01	-	
Absolute Surface Roughness	$\epsilon =$	4.57 E-05	m	0.00015 ft
Slit depth		1.10 E-02	m	0.433 in
	$\Delta P =$	1.59 E+05	N/m ²	23 psf
	$\rho =$	1.04 E+03	kg/m ³	64.9 lb/ft ³
Velocity	$U_1 = C_v \sqrt{2 \frac{\Delta P}{\rho}} =$	1.75 E+01	m/s	57.41 ft/s
	Viscosity $\mu =$	1.00 E-03	Pa/s	1 centipoise
	Kinematic Viscosity $\nu_L = \mu/\rho =$	9.62 E-07	m ² /s	3.01 E-6 ft ² /s
	slit length $l =$	7.62 E-02	m	3 in

The respirable portion was calculated using Saunter Mean Diameter

$$SMD = \frac{500 d_o^{1.2} \nu_L^{0.2}}{U_L} \quad \text{and the Rosin-Rammler drop size distribution}$$

$$Q = 1 - e^{-(D_p/X)^q}$$

SMD using the Rosin Rammler drop size distribution

$$\frac{SMD}{X} = \left[\Gamma \left(1 - \frac{1}{q} \right) \right]^{-1}$$

Where: Γ denotes the gamma function

To calculate the Saunter Mean Diameter (SMD) and drop size distribution the following variable constants were assumed. The value of q and the corresponding value of SMD/X were taken from Page 3 of WHC-SD-GN-TI-30003

Respirable drop size	$D_p =$	20	μm	7.87 E-4	in
A fitting Constant	$q =$	2.4			
Based on q	$SMD/X =$	0.65415			

The calculation sequence giving the equations in the sequence used and the final result follows.

slit width w	$=$	1.943 E-04	m	7.65 E-03	in
Equivalent diameter $2^*w = d_e$	$=$	3.89 E-04	m	1.53 E-2	in
depth/ d_e	$=$	2.83 E+01			
ϵ/d_e	$=$	1.18 E-01			
Estimated Reynolds number R_e	$=$	3.25 E+03			
$\frac{1}{\sqrt{f_d}} = -2 \log_{10} \left(\frac{\epsilon/d_e}{3.7} + \frac{2.51}{R_e \sqrt{f_d}} \right)$ estimated f_d	$=$	1.16 E-01			
$\epsilon/d_e/3.7$	$=$	3.18 E-02			
$2.51/R_e * f^{1/2}$	$=$	2.27 E-03			
$-2 * \log(\epsilon/d_e/3.7 + 2.51/R_e * f^{1/2})$	$=$	2.94 E+00			
f_d	$=$	1.16 E-01			
K_f	$=$	3.28 E+00			
$K + 1 = K_f + K_v + 1$	$=$	4.77 E+00			
Liquid jet velocity $U_L = \sqrt{\frac{2 \Delta P}{\rho (K + 1)}}$	$=$	8.00 E+00	m/s	62.2	ft/s
Reynolds number $R_e = \frac{D_e U_L \rho}{\mu}$	$=$	3.23 E+03			
$SMD = \frac{500 d_p^{1.7} v_{L,2}^{0.3}}{U_L}$	$=$	3.16 E+02	μm	1.24 E-2	in
$Q = 1 - \exp\left(-\left(\frac{SMD}{200}\right)^2\right)$	$=$	4.79 E-04			
Total spray flow rate $= N_t = C_v A U_L$	$=$	1.18 E-04	m ³ /s	3.12 E-2	gal/s
Release flow rate $= N_p = Q N_t$	$=$	5.68 E-08	m ³ /s	1.58 E-5	gal/s
Total22 volume released $= 150 \text{ min} * 60 \text{ s} * U_t$	$=$	1.07 E+00	kL	2.75 E+02	gal
Volume released $= 150 \text{ min} * 60 \text{ s} * U_p$	$=$	5.11 E-04	kL	1.32 E-01	gal

APPENDIX C

**GENII DOSE CONSEQUENCE CALCULATION
FOR THE 340 FACILITY**

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From: Criticality & Radiological Analyses
Phone: 376-9710 H4-64
Date: September 19, 1994
Subject: GENII DOSE CONSEQUENCE CALCULATION FOR THE 340 FACILITY

LVN-8D150-94-013

To: D. J. Johnson H4-70

cc: J. S. Davis H4-64
J. P. Estrellado, Jr. H4-70
B. E. Hey H4-64
J. C. Van Keuren H4-64
J. W. Daughtry H4-64
LVN File/LB

As requested, radiological dose consequence calculations for the 340 facility have been completed. Doses were calculated for a release of 1 Ci of the isotopes specified by the customer. The computer codes used in this analysis were GENII version 1.485 and GXQ Version 3.1. The results are reported in the attachment.

If you have any questions, please call me on 376-9710.

Loc V. Nguyen
Engineer

siw

Attachment

Concurrence:

J. W. Daughtry, Manager
Criticality & Radiological Analyses

9-20-94

Date

GENII DOSE CONSEQUENCE CALCULATION FOR THE 340 FACILITY

L. V. Nguyen
September 20, 1994

PURPOSE

Potential maximum individual dose consequences to the onsite 100 meter and site boundary receptors from the postulated accident involving the 340 facility radioactive inventory have been calculated. The accident was evaluated as a ground-level release.

SOURCE TERMS

The source terms were 1 Ci of each isotope as specified by the customer in Appendix A. The release was assumed to be in the form of oxide.

METHODOLOGIES AND ASSUMPTIONS

Acute maximum individual ground level release 99.5 percentile dispersion factors (X/Q) were generated using GXQ Version 3.1 (B. E. Hey, 1993) for 16 onsite 100-meter and site boundary locations. In addition, the onsite 100-meter and site boundary overall average 95 percentile X/Qs were determined using GXQ to compare with the maximum sector individual X/Q. The higher X/Q will be used for dose calculations (NRC Regulatory Guide 1.145).

RECEPTOR DESCRIPTION

- Onsite: For the ground level releases, the receptor is normally 100 m in the worst direction (WHC-CM-4-46). Doses calculated for the onsite receptor include inhalation and submersion. The onsite maximum individual X/Q at 100 meters was calculated for 16 sectors. The maximum onsite individual for acute ground level release from the 340 Facility was calculated to be located at 100 meters north, with an acute 99.5 percentile X/Q of 4.21 E-02 s/m^3 . The 100 meter X/Q for all 16 sectors are listed in Table 1. The asterisk in Table 1 identifies the worst onsite X/Q.
- Site Boundary: The site boundary receptor is located at the site boundary in the worst direction. Where the site is bounded by the Columbia River, the site boundary is taken to be at the nearer bank of the river for maximum individual doses. This receptor is assumed to stay at this location for the duration of the accident. Doses calculated include inhalation and submersion. The maximum offsite individual 99.5 percentile X/Q for an acute ground level release from the 340 Facility is 3.77 E-03 s/m^3 (345 meters E). The asterisk in Table 2 identifies the worst site boundary X/Q.

CODE DOCUMENTATION

- GENII version 1.485 (12/3/90)
- GENII Default Parameter Values (28-Mar-90 RAP)
- Radionuclide Master Library (7/23/93 PDR)
- PNL Food Transfer Factor Library (7/19/93)
- External Dose Factor Library (8-May-90-RAP)
- Internal Dose Increments, PNL Solubilities, (7/23/93 PDR)
- Joint Frequency Data: 300 Area, 10 m, Pasquill A-G (1983-1991 Average)
- GXQ Version 3.1 (1993)

A typical GENII and GXQ input files are attached as Appendix B and C.

RESULTS

The resulting effective dose equivalents (EDE) and limiting organ doses for onsite and offsite receptors calculated by GENII with 99.5 percentile meteorology are shown in Tables 3 and 4. Inhalation doses were 50-year committed doses based on an acute uptake. Submersion doses were calculated using a semi-infinite cloud model.

Any external dose due to plume submersion was included in both the EDE and organ dose for GENII calculations. Building wake and plume meander were not taken into account as requested by the customer.

NOTE: *Please, be aware of the fact that the new joint frequency data, and HRC Regulatory Guide 1.145 (Ref. 4) atmospheric dispersion coefficient (X/Q) methods were applied for dose calculations. These changes may affect consistency with previous X/Qs and dose consequences. This document is based on the best available information at this time.*

Note: When using the unit release consequences presented in this document, note that (a) release fractions considering only respirable size particles may not be appropriate for determining source term ratios, (b) scaling of unit release consequences may not be appropriate for changes other than source term quantity, and (c) the technical peer review for unit release consequences does not cover consequences scaled from the unit release consequences. Scaled consequences must also be reviewed by Radiological & Toxicological Analysis or their designees, as required in SA&E Work Procedure 6.2 (WHC CM-6-32). In general, the HEDOP review of the original analysis is sufficient.

Table 1: Acute 99.5 Percentile X/Qs in 16 Sectors, and 95 Percentile Overall Average X/Q for 100 Meters, all for Ground Level Releases.

Sector	Dispersion Factors X/Q (s/m ³)
S	3.18 E-02
SSW	1.16 E-02
SW	4.39 E-03
WSW	3.88 E-03
W	1.51 E-02
WNW	2.90 E-02
NW	3.28 E-02
NNW	3.29 E-02
N	4.21 E-02
NNE	2.89 E-02
NE	2.93 E-02
ENE	1.60 E-02
E	3.16 E-02
ESE	2.99 E-02
SE	3.16 E-02
SSE	3.23 E-02
OVERALL 95 PERCENTILE X/Q	3.44 E-02

Table 2: Acute 99.5 Percentile X/Qs at Site Boundary Locations, for Ground Level Release from 340 Facility.

Sector	Distance (m)	Dispersion Factors X/Q (g/m^3)
S	6475	4.25 E-05
SSW	2590	5.36 E-05
SW	3515	1.32 E-05
WSW	3515	1.01 E-05
W	6375	1.88 E-05
WNW	28675	6.02 E-06
NW	49025	3.54 E-06
NNW	51800	3.33 E-06
N	17020	1.65 E-05
NNE	505	1.83 E-03
NE	505	1.85 E-03
ENE	345	1.90 E-03
E	345	3.77 E-03
ESE	345	3.58 E-03
SE	535	1.81 E-03
SSE	830	8.89 E-04
Overall sector 95 percentile		1.51 E-03

Table 3. Maximum Individual Doses for Onsite 100
Meter Receptor Due to an Aerosol Release
of 1 Ci of Each Isotope from 340 Facility

Radionuclide	External Dose (rem)	Annual Effective Dose Equivalent (rem)
MN 54	7.0 E-03	9.6 E-02
CO 60	2.1 E-02	9.6 E+00
SR 90	1.7 E-06	2.8 E+00
Y 90	6.1 E-05	1.2 E-01
ZR 95	6.3 E-03	2.2 E-01
NB 95	7.0 E-03	8.7 E-02
TC 99	3.2 E-07	1.3 E-01
RU 106	2.0 E-03	6.6 E+00
SB 125	4.0 E-03	1.8 E-01
CS 134	1.4 E-02	6.1 E-01
CS 137	4.7 E-03	4.3 E-01
CE 144	1.2 E-04	5.2 E+00
EU 152	1.0 E-02	2.9 E+00
EU 154	1.0 E-02	3.9 E+00
EU 155	3.5 E-04	5.7 E-01
TA 182	1.1 E-02	6.2 E-01
U 234	1.7 E-06	1.9 E-03
U 235	0.0 E+00	1.7 E+03
U 236	1.3 E-06	1.8 E+03
PU 238	1.3 E-06	4.0 E+03
U 238	1.2 E-06	1.7 E+03
PU 239	8.8 E-07	4.3 E+03
PU 240	1.3 E-06	4.2 E+03
AM 241	9.0 E-05	6.2 E+03
PU 241	3.1 E-12	6.9 E+01
CN 242	1.3 E-06	2.4 E+02
PU 242	0.0 E+00	4.1 E+03

Table 4. Maximum Individual Doses Calculated for Site Boundary
Receptor at 345 m E, Due to An Aerosol Release
of 1 Ci of Each Isotope from 340 Facility

Radionuclide	External Dose (rem)	Annual Effective Dose Equivalent (rem)
MN 54	6.5 E-04	8.3 E-03
CO 60	1.9 E-03	2.4 E-01
SR 90	1.6 E-07	2.4 E-01
Y 90	5.6 E-06	1.0 E-02
ZR 95	5.8 E-04	1.9 E-02
NB 95	6.5 E-04	7.5 E-03
TC 99	2.9 E-08	1.1 E-02
RU 106	1.8 E-04	5.6 E-01
SB 125	3.7 E-04	1.6 E-02
CS 134	1.3 E-03	5.2 E-02
CS 137	4.4 E-04	3.6 E-02
CE 144	1.1 E-05	4.5 E-01
EU 152	9.2 E-04	2.5 E-01
EU 154	9.4 E-04	3.3 E-01
EU 155	3.2 E-05	4.9 E-02
TA 182	1.0 E-03	5.3 E-02
U 234	1.6 E-07	1.6 E-02
U 235	0.0 E+00	1.5 E-02
U 236	1.2 E-07	1.5 E-02
U 238	1.1 E-07	1.4 E-02
PU 239	8.1 E-08	3.6 E-02
PU 240	1.2 E-07	3.6 E-02
AM 241	2.9 E-13	5.9 E+00
PU 241	8.3 E-06	5.3 E-02
CM 242	1.2 E-07	2.0 E-01
PU 242	0.0 E+00	3.5 E-02

REFERENCES

- Napier, B. A., et al., 1988, *GENII - The Hanford Environmental Radiation Dosimetry Software System*, PNL-6484, Pacific Northwest Laboratory, Richland, Washington.
- Hey, B. E., 1993, *GXQ Program Users' Guide*, WHC-SD-GN-SWD-30002, Westinghouse Hanford Company, Richland, Washington.
- WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, Westinghouse Hanford Company, Richland, Washington.
- U.S. Nuclear Regulatory Commission, November 1982, *Atmospheric Dispersion Models For Potential Accident Consequence Assessments At Nuclear Power Plants*, Regulatory Guide 1.145, Revision 1, Office Of Nuclear Regulatory Research.

APPENDIX A

CC: MAIL FROM D. J. JOHNSON TO J. C. VAN KEUREN

DATE: September 14, 1994

340 Vault tanks expected isotopes	
Isotopes	CI
^{54}Mn	1.0
^{60}Co	1.0
^{90}Sr	1.0
^{95}Zr	1.0
^{95}Nb	1.0
^{99}Tc	1.0
^{106}Ru	1.0
^{125}Sb	1.0
^{134}Cs	1.0
^{137}Cs	1.0
^{144}Ce	1.0
^{152}Eu	1.0
^{154}Eu	1.0
^{155}Eu	1.0
^{182}Ta	1.0
^{234}U	1.0
^{235}U	1.0
^{236}U	1.0
^{238}U	1.0
^{238}Pu	1.0
$^{239}/^{240}\text{Pu}$	1.0
^{240}Pu	1.0
^{241}Pu	1.0
^{242}Pu	1.0
^{241}Am	1.0
$^{242}/^{243}\text{Cm}$	1.0

APPENDIX B
SAMPLE GENII INPUT DECK

ONSITE 100 METERS ACUTE GROUND LEVEL INDIVIDUAL RELEASE

Program GENII Input File ##### 8 Jul 88 ###
 Title: ACUTE ONSITE INDIVIDUAL DOSES FOR X/Q OF 4.21 E-02 RELEASE

\SAMPL\G-AIR.AC

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
 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-----
 T Report AEDE only F Drinking water ingestion 7,8
 T Report by radionuclide F Aquatic foods ingestion 7,8
 T Report by exposure pathway F Terrestrial foods ingestion 7,9
 F Debug report on screen F Animal product ingestion 7,10
 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

Use when	Release Terms transport selected			Basic Concentrations near-field scenario, optionally				
	Air	Surface	Buried	Air	Surface	Deep	Ground	Surface
Release Radio- nuclide	/yr	Water /yr	Waste /m3	/m3	Soil /unit	Soil /m3	Water /L	Water /L
MN54	1.0							
CO60	1.0							
AM241	1.0							
CS137	1.0							
CS134	1.0							
CE144	1.0							
CM242	1.0							
EU152	1.0							
EU154	1.0							
EU155	1.0							
SR90	1.0							
Y 90	1.0							
ZR95	1.0							
NB95	1.0							
TC99	1.0							
RU106	1.0							

SB125 1.0
 TA182 1.0
 U 234 1.0
 U 235 1.0
 U 236 1.0
 U 238 1.0
 PU238 1.0
 PU239 1.0
 PU240 1.0
 PU241 1.0
 PU242 1.0

-----Derived Concentrations-----				
Use when	measured values are known			
Release Radio- nuclide	Terres.	Animal	Drink	Aquatic
	Plant	Product	Water	Food
	/kg	/kg	/L	/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 #####

0 Prior to the beginning of the intake period: (yr)
 0 When was the inventory disposed? (Package degradation starts)
 0 When was LIOC? (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)
1	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)
4.21E-2	Chi/Q or PM value	0	Stack radius (m)
0	MI sector index (1-S)	0	Effluent temp. (C)
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)

```

=====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 withdrawal 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

=====SECTION 3=====
0  Waste form/package half life, (yr)
0  Waste thickness, (m)
0  Depth of soil overburden, m

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

```

EXPOSURE

```

=====SECTION 5=====
0  Exposure time:
0  Plume (hr)
0  Soil contamination (hr)
0  Swimming (hr)
0  Boating (hr)
0  Shoreline activities (hr)
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)

=====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)

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

#####

APPENDIX C

SAMPLE ONSITE 100 METERS GXQ INPUT DECK

ONSITE 100 METER 99.5 PERCENTILE X/Q

OXO Version 3.1.8
June 6, 1993

General Purpose Atmospheric Dispersion Code
Produced by Radiological & Toxicological Analytic
Westinghouse Hanford Company

User's Guide documented in WUC-SD-OR-SMO-3102 Rev. 0.
Validation documented in WUC-SD-OR-SMO-3103 Rev. 0.
Code Custodian is Brit E. May, WMC, ext. 6-2921.

Run Date = 2/ 1/1994
Run Time = 10:44 24.03

INPUT Echo:

Input title line
c OXO Ver. 3.18 Input file
c mode

c MODE CHOICES:

c mode = 1 then X/Q based on Hanford site specific meteorology
c mode = 2 then X/Q based on atmospheric stability class and wind speed
c mode = 3 then X/Q plot file is created

c LOGICAL CHOICES:

c ftox inorm fcdt fchk fsite fpop fcon
c ftox = f then joint frequency used to compute frequency to exceed X/Q
c fcdt = f then joint frequency used to compute annual average X/Q
c fchk = f then joint frequency data is normalized (as in GENII)
c fcdt = f then joint frequency data is unnormalized
c fchk = f then no annual distribution file created (CDF OUT)
c fchk = f then X/Q parameter print option turned on
c fsite = f then X/Q based on joint frequency data for all 16 sectors
c fpop = f then X/Q is population weighted
c fcon = f then X/Q is air concentration
c fcon = f then X/Q is integrated exposure

c MODEL CHOICES:

c fdep fstack fpm frite fgrav fush fflow fwind
c fdep = 1 then plume depletion model turned on (Chamberlain model)
c fstack = 1 then HRC RG 1.145 building wake model turned on
c fpm = 1 then MACS virtual distance building wake model turned on
c fpm = 2 then HRC RG 1.145 plume meander model turned on
c frite = 1 then 5th Power Law plume meander model turned on
c frite = 1 then momentum/buoyancy plume rise model turned on
c frite = 2 then momentum/buoyancy plume rise model turned on, buoyancy
c frite = 3 rise based on sensible heat emission
c fgrav = 1 then gravity settling model turned on
c fgrav = 1 then gravity settling model turned on
c fush = 1 then stack downwash model turned on
c fflow = 1 then slopes adjusted for volume flow rate
c fwind = 1 then wind speed corrected for plume height

```

c      = 0 to turn any of the above models off
c
c PARAMETER INPUT:
c stack      wind      frequency
c release    speed     to
c height     height    exceed
c (m)        (m)       (m)
c 0.00000E+00 1.00000E+01 1.00000E+03 5.00000E-01 1.00000E+00
c
c building   building   release   deposition   gravitational
c width      height     duration   velocity     settling
c (m)        (m)        (hr)      (m/s)        velocity
c 1.52400E+01 7.80000E+00 2.00000E+00 1.00000E-03 1.00000E-03
c
c          initial      sensible
c          plume        heat
c plume     flow        stack   emission
c density   rate        diameter rate
c (g/cc)    (m3/s)      (m)     (cal/s)
c 1.20000E-03 0.00000E+00 5.00000E+00 4.18000E+05
c
c RECEPTOR DEPENDENT DATA
c FOR MODE make RECEPTOR DEPENDENT DATA
c 1 (site specific) sector distance z-height
c 2 (by class & wind speed) class windspeed distance offset z-height
c 3 (create plot file) class windspeed xmax ymax xmin xqmin power
c
c RECEPTOR PARAMETER DESCRIPTION:
c sector = 0, 1, 2... (all, S, SSW, etc.)
c distance = meters
c class = 1, 2, 3, 4, 5, 6, 7 (P-G stability class A, B, C, D, E, F, G)
c windspeed = m/s
c offset = meters offset from plume centerline
c xmax = maximum distance to plot or calculate to (m)
c imax = distance intervals
c ymax = maximum offset to plot (m)
c jmax = offset intervals
c xqmin = minimum scaled X/Q to calculate
c power = exponent in power function step size
c
MODE:
Site specific X/Q calculated

```

```

LOGICAL CHOICES:
Joint frequency used to calculate X/Q based on
frequency of exceedance.
No normalization of joint frequency.
X/Q calculated for single sector.
Output is atmospheric dispersion coefficient.

```

MODELS SELECTED:

```

WARNING/ERROR MESSAGES:
WARNING #3 - Scaled X/Q units (shown as s/m3 or 1/m3
below) do not reflect the user specified
scaling factor.

```

```

JOINT FREQUENCY DATA:
300 AREA - 10 M - Pasquill A - G (1983 - 1991 Average)
Created 8/26/92 KR

```

Input title line

	TOTAL	AVERAGE	
RECEPT SECT.	POPULATION	INDIVIDUAL	
	SCALED	SCALED	ATM. WIND

SECTOR	DISTANCE (m)	HEIGHT (m)	FREQ. (X)	POPULATION	X/Q (s/m3)	X/Q (s/m3)	STAR. CLASS	SPEED (m/s)
S	100	0	11.55	1	3.18E-02	3.18E-02	F	.89
SSW	100	0	5.07	1	1.16E-02	1.16E-02	F	2.65
SW	100	0	2.39	1	4.39E-03	4.39E-03	F	7.15
WSW	100	0	1.82	1	3.88E-03	3.88E-03	C	.89
W	100	0	4.20	1	1.51E-02	1.51E-02	E	.89
WNW	100	0	6.49	1	2.90E-02	2.90E-02	G	2.65
NW	100	0	8.60	1	3.28E-02	3.28E-02	F	.89
NNW	100	0	5.97	1	3.29E-02	3.29E-02	F	.89
N	100	0	8.76	1	4.21E-02	4.21E-02	F	.89
NNE	100	0	8.50	1	2.89E-02	2.89E-02	G	2.65
NE	100	0	9.64	1	2.93E-02	2.93E-02	G	2.65
ENE	100	0	6.13	1	1.60E-02	1.60E-02	G	4.70
E	100	0	4.55	1	3.16E-02	3.16E-02	F	.89
ESE	100	0	2.86	1	2.99E-02	2.99E-02	G	2.65
SE	100	0	5.20	1	3.16E-02	3.16E-02	F	.89
SSE	100	0	7.95	1	3.23E-02	3.23E-02	F	.89

Execution Time = 0 hr 0 min 5.60 sec

Stop - Program terminated.

OFFSITE RECEPTORS 99.3 PERCENTILE X/Q

GXQ Version 3.1 B
June 8, 1993

General Purpose Atmospheric Dispersion Code
Produced by Radiological & Toxicological Analysis
Westinghouse Hanford Company

Users Guide documented in WMC-SD-GN-SMD-3002 Rev. 0.
Validation documented in WMC-SD-GN-SMD-3003 Rev. 0.
Code Custodian is Brit E. Key, WMC, ext. 8-2921.

Run Date = 9/14/1994
Run Time = 14:24 44.26

INPUT ECHO:

GXQ test input file
c GXQ Ver. 3.1 Input File
c mode
1
c
c MODE CHOICE:
c mode = 1 then X/Q based on Hanford site specific meteorology
c mode = 2 then X/Q based on atmospheric stability class and wind speed
c mode = 3 then X/Q plot file is created
c
c LOGICAL CHOICES:
c ifox inorm icdf ichk isite ipop icon
T F F F F F F F
c ifox = t then joint frequency used to compute frequency to exceed X/Q
c = f then joint frequency used to compute annual average X/Q
c inorm = t then joint frequency data is normalized (as in GENII) f
c = f then joint frequency data is unnormalized
c icdf = t then cumulative distribution file created (CDF.OUT)
c = f then no cumulative distribution file created
c ichk = t then X/Q parameter print option turned on
c = f then no parameter print
c isite = t then X/Q based on joint frequency data for all 16 sectors
c = f then X/Q based on joint frequency data of individual sectors
c ipop = t then X/Q is population weighted
c = f then no population weighting
c icon = t then X/Q is air concentration
c = f then X/Q is integrated exposure
c
c MODEL CHOICES:
c iddep iwake ipm irise igrav iwash iflow iwind
0 0 0 0 0 0 0 0
c iddep = 1 then plume depletion model turned on (Chamberlain model)
c iwake = 1 then NRC RG 1.145 building wake model turned on
c = 2 then MACCS virtual distance building wake model turned on
c ipm = 1 then NRC RG 1.145 plume meander model turned on
c = 2 then 5th Power Law plume meander model turned on
c = 3 then sector average model turned on
c irise = 1 then momentum/buoyancy plume rise model turned on, buoyancy
c rise based on sensible heat emission
c = 2 then momentum/buoyancy plume rise model turned on, buoyancy
c rise based on initial plume density
c igrav = 1 then gravitational settling model turned on
c iwash = 1 then stack downwash model turned on
c iflow = 1 then sigma adjusted for volume flow rate
c iwind = 1 then wind speed corrected for plume height
c = 0 to turn any of the above models off
c

```

c PARAMETER INPUT:
c stack      wind      frequency
c release    speed     to
c height     height    exceed
c (m)        (m)       (m)
c 0.00000E+00 1.00000E+00 1.00000E+03 5.00000E-01 1.00000E+00
c
c building   building   release    deposition   gravitational
c width      height     duration   velocity    settling
c (m)        (m)       (hr)      (m/s)       velocity
c 0.00000E+00 0.00000E+00 1.00000E+00 1.00000E-03 1.00000E-03
c
c initial    initial    sensible
c plume      plume      heat
c density    flow       emission
c (g/cc)     rate       diameter
c 1.22000E-03 1.00000E+00 1.00000E+00 4.18000E+05
c
c RECEPTOR DEPENDENT DATA
c FOR MODE   make      RECEPTOR DEPENDENT DATA
c 1 (site specific) sector distance z-height
c 2 (by class & wind speed) class windspeed distance offset z-height
c 3 (create plot file) class windspeed xmax imax ymax jmax xqmin power
c
c RECEPTOR PARAMETER DESCRIPTION:
c sector = 0, 1, 2... (all, S, SSW, etc.)
c distance = meters
c class = 1, 2, 3, 4, 5, 6, 7 (P-G stability class A, B, C, D, E, F, G)
c windspeed = m/s
c offset = meters offset from plume centerline
c xmax = maximum distance to plot or calculate to (m)
c imax = distance intervals
c ymax = maximum offset to plot (m)
c jmax = offset intervals
c xqmin = minimum scaled X/Q to calculate
c power = exponent in power function step size

MODE:
Site specific X/Q calculated

LOGICAL CHOICES:
Joint frequency used to calculate X/Q based on
frequency of exceedance.
No normalization of joint frequency.
X/Q calculated for single sector.
Output is atmospheric dispersion coefficient.

MODELS SELECTED:

WARNING/ERROR MESSAGES:
WARNING #3 - Scaled X/Q units (shown as s/m3 or 1/m3
below) do not reflect the user specified
scaling factor.

JOINT FREQUENCY DATA:
300 AREA - 10 M - Pasquill A - G (1983 - 1991 Average)
Created 8/26/92 KR

GIX test input file

```

SECTOR	DISTANCE (m)	RECEIPT HEIGHT (m)	SECT. FREQ. (%)	POPULATION	TOTAL POPULATION X/Q (s/m3)	AVERAGE INDIVIDUAL X/Q (s/m3)	ATM. STAB. CLASS	WIND SPEED (m/s)
S	6475	0	11.55	1	4.25E-05	4.25E-05	F	.89
SSW	2590	0	5.07	1	5.36E-05	5.36E-05	F	2.65
SW	3515	0	2.39	1	1.32E-05	1.32E-05	F	7.15
WSW	3515	0	1.82	1	1.01E-05	1.01E-05	E	4.70
W	6575	0	4.20	1	1.88E-05	1.88E-05	E	.89
WNW	28675	0	6.49	1	6.02E-06	6.02E-06	G	2.65
NW	49025	0	8.60	1	3.54E-06	3.54E-06	F	.89
NNW	51800	0	5.97	1	3.33E-06	3.33E-06	F	.89
N	17020	0	8.76	1	1.65E-05	1.65E-05	F	.89
NNE	505	0	8.50	1	1.83E-03	1.83E-03	G	2.65
NE	505	0	9.64	1	1.85E-03	1.85E-03	G	2.65
ENE	345	0	6.13	1	1.90E-03	1.90E-03	G	4.70
E	345	0	4.55	1	3.77E-03	3.77E-03	G	2.65
ESE	345	0	2.86	1	3.58E-03	3.58E-03	G	2.65
SE	535	0	5.20	1	1.81E-03	1.81E-03	F	.89
SSE	830	0	7.95	1	8.89E-04	8.89E-04	F	.89

Execution Time = 0 hr 0 min 5.66 sec

Stop - Program terminated.

CHECKLIST FOR PEER REVIEW

Document Reviewed: Loc V. Nguyen, "GENII DOSE CONSEQUENCE CALCULATION FOR THE 340 FACILITY," July 14, 1994.

Scope of Review: Entire Document

Yes	No	NA	
☐	☐	☒	* Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
☒	☐	☐	Problem completely defined.
☒	☐	☐	Accident scenarios developed in a clear and logical manner.
☒	☐	☐	Necessary assumptions explicitly stated and supported.
☒	☐	☐	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☒	☐	☐	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☒	☐	☐	Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.
☒	☐	☐	Software input correct and consistent with document reviewed.
☒	☐	☐	Software output consistent with input and with results reported in document reviewed.
☐	☐	☒	Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
☐	☐	☒	Safety margins consistent with good engineering practices.
☐	☐	☒	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.
☐	☐	☒	Format consistent with appropriate NRC Regulatory Guide or other standards
☐	☒	*	Review calculations, comments, and/or notes are attached.
☒	☐	☐	Document approved. <i>B. E. Hen</i> 10/15/94
<i>B. E. Hen</i>			7/14/94
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.

☒ ☐ ☐ Analysis entered into analysis database

Loc V. Nguyen *Loc Nguyen* July 14, 1994
(Printed Name and Signature) Date

HEDOP REVIEW CHECKLIST
for
Radiological and Nonradiological Release Calculations

Loc V. Nguyen, "GENII DOSE CONSEQUENCE CALCULATION FOR THE 340 FACILITY,"
September 14, 1994. *Memo, L.V. Nguyen to D.J. Johnson*

Submitted by: L. V. Nguyen.

Date Submitted: September 14, 1994

Scope of Review: Entire Document

YES NO* N/A

☒	☐	☐	1. A detailed technical review and approval of the environmental transport and dose calculation portion of the analysis has been performed and documented.
☐	☐	☒	2. Detailed technical review(s) and approval(s) of scenario and release determinations have been performed and documented.
☒	☐	☐	3. HEDOP-approved code(s) were used.
☒	☐	☐	4. Receptor locations were selected according to HEDOP recommendations.
☒	☐	☐	5. All applicable environmental pathways and code options were included and are appropriate for the calculations.
☒	☐	☐	6. Hanford site data were used.
☐	☐	☒	7. Model adjustments external to the computer program were justified and performed correctly.
☒	☐	☐	8. The analysis is consistent with HEDOP recommendations.
☐	☐	☒	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.)
☒	☐	☐	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 9/16/94
HEDOP-Approved Reviewer (Printed Name and Signature) Date

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

3. GENII (HEDOP-approved) was used. GXQ (not HEDOP-approved) was used to calculate X/Qs because GENII does not calculate X/Qs in accordance with NRC Reg. 23 Guide 1.145.
6. Hanford site joint frequency meteorology data were used in GXQ.

APPENDIX D
HAZARDOUS MATERIAL RELEASE ESTIMATES

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HAZARDOUS MATERIAL RELEASE ESTIMATES

A review of the RLWS Disposal Approval Requests prepared by PNL identified the types of hazardous chemicals received at the 340 Facility. A majority of the waste, over 93%, is water. The chemicals appearing most frequently in the waste are:

Nitric Acid
Sodium Hydroxide
Sodium Nitrate
UltimaGold (solvent Di-isopropyl-nathalene)
Optifluor (solvent alkylbenzene)
Sodium Phosphate
Potassium Persulfate
Sodium Carbonate
Potassium Nitrate
Potassium Hydroxide
Hydrochloric Acid
Sulfuric Acid
Ammonium Hydroxide
Oxalic Acid
Dimethylsulfoxide

PNL uses sodium hydroxide to neutralize nitric acid forming sodium nitrate, the predominate contaminant. This agrees with the 340 Facility Vault Tank sample data shown in Table 2-2.

Using nitrate as the worst-case chemical, the estimated spray release airborne concentrations become:

$$\text{Spray Release (mg/m}^3\text{)} = V \text{ (m}^3\text{/s)} \times X/Q \text{ (s/m}^3\text{)} \times \text{source (mg/m}^3\text{)}$$

where

$$\begin{array}{lll} V & = 5.68 \times 10^{-8} \text{ m}^3/\text{s} & \text{Appendix B} \\ X/Q \text{ onsite} & = 4.21 \times 10^{-2} \text{ s/m}^3 & \text{Appendix C} \\ X/Q \text{ offsite} & = 3.77 \times 10^{-3} \text{ s/m}^3 & \text{Appendix C} \\ \text{source} & = 3.42 \times 10^7 \text{ mg/m}^3 & 3.42 \times 10^4 \text{ mg/L in Table 2-2} \end{array}$$

Assuming the source to be NaNO_3 , the source value becomes:

$$3.42 \times 10^7 \times \frac{85 \text{ g NaNO}_3}{65 \text{ g NO}_3} = 4.47 \times 10^7 \text{ mg/m}^3$$

Estimated onsite release concentration:

$$0.11 \text{ mg/m}^3 = 5.68 \times 10^{-8} \text{ m}^3/\text{s} \times 4.21 \times 10^{-2} \text{ s/m}^3 \times 4.47 \times 10^7 \text{ mg/m}^3$$

Estimated offsite release concentration:

$$9.6 \times 10^{-3} \text{ mg/m}^3 = 5.68 \times 10^{-8} \text{ m}^3/\text{s} \times 3.77 \times 10^{-3} \text{ s/m}^3 \times 4.47 \times 10^7 \text{ mg/m}^3$$

The Hanford Emergency Response Planning Guidelines for chemicals are intended for use in performing site safety classification, hazard evaluations, and emergency preparedness planning (WHC-CM-4-46, Appendix D). The risk-acceptance guidelines for onsite exposure recommends using the ERPG-2 to ERPG-3 values over the frequency range of 1 to 10^{-6} . For offsite, risk-acceptance guidelines recommend using ERPG-1 to ERPG-2 values over the frequency range of 1 to 10^{-6} . The most restrictive range is ERPG-1 to ERPG-2. ERPG values do not exist for every chemical and WHC-CM-4-46 recommends using threshold limit value - time weighted average (TLV-TWA) values when no ERPG values can be found. Neither ERPG nor TLV-TWA values could be found for NaNO₂, using the Hanford information computer listing of ERPG values, the National Institute for Occupational Safety and Health Pocket Guide to Chemical Hazards or the American Conference of Governmental Industrial Hygienists pocket guide for Threshold Limit Values and Biological Exposure Indices. However, using the ERPG limits for a more hazardous chemical provides a greater degree of conservatism and further reduces the risk of adverse effects on the general population. The chemical received at the 340 Facility that presents a potential hazard to personnel, the environment, and the public is sodium hydroxide. The PNL facilities adjust the radioactive liquid waste to a pH of 10, using sodium hydroxide to reduce corrosion of the stainless steel. The ERPG (Hanford Info) sodium hydroxide values for the comparison are:

Sodium Hydroxide Onsite (mg/m ³)	ERPG-2 40	to	ERPG-3 100	Estimated Value 0.11
Sodium Hydroxide Offsite (mg/m ³)	ERPG-1 2	to	ERPG-2 40	Estimated Value 0.0096

These values show that comparing the chemical having the highest concentration with the chemical that is a potential hazard for 340 Facility personnel gives values far below the Emergency Response Planning Guidelines. Based on this comparison, releases for the other hazardous chemicals are expected to be well within the risk-acceptance guidelines.

APPENDIX E
APPROVAL OF INTERIM SAFETY BASIS (ISB)
FOR THE 340 WASTE HANDLING FACILITY

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HNF-SD-WM-ISB-003, REV 2

Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

SEP 23 1996

96-LEP-089

Dr. A. L. Trego, President
Westinghouse Hanford Company
Richland, Washington

Dear Dr. Trego:

APPROVAL OF INTERIM SAFETY BASIS (ISB) FOR THE 340 WASTE HANDLING FACILITY

- References:
- 1) Letter, A. J. Diliberto, WHC, to T. K. Teynor, RL, "340 Waste Handling Facility Status of Annual Review of the Interim Safety Basis and Unreviewed Safety Question Evaluations," 9651506, dated April 19, 1996.
 - 2) Letter, T. K. Teynor, to President, WHC, "Approval of the 340 Waste Handling Facility Interim Safety Basis", 95-LEP-037, dated June 9, 1995.

Department of Energy, Richland Operations Office (RL); has completed the review of the 340 Waste Handling Facility ISB (WHC-SD-WM-ISB-003, Rev. 1). All of the reviewers comments have been satisfactorily resolved and dispositioned. RL's review and conclusion is documented in the attached Safety Evaluation Report (SER). The ISB presents an adequate and acceptable base for authorization of the analyzed operations at the 340 Waste Handling Facility. This SER is to be considered part of the safety and authorization basis for operations at the 340 Waste Handling Facility.

Upon full revision and implementation, this ISB presented in Reference 1 will replace the approval to operate under ISB Rev. 0. as stated in Reference 2. ISB Rev 1. will be the Authorization Basis for the 340 Waste Handling Facility.

A. L. Trego, President
96-LEP-089

-2-

If you have any questions, you may contact me on 376-7395, or your staff may contact Anna Beard, on 376-7472.

Sincerely,

John D. Wagoner
Manager

WPD:AVB

Attachment

cc: w/attach

A. J. Diliberto, WHC

D. L. Halgren, WHC

H. J. Hatch, FDH

L. W. Roberts, WHC

**Safety Evaluation Report (SER) for the
340 Waste Handling Facility (WHC-SD-
WM-ISC-003, Rev 1)**

August, 1996

**U. S. Department of Energy, Richland Operations
Office, Office of Waste Management**

340 ISB SER

August, 1996

EXECUTIVE SUMMARY

The Safety Evaluation Report (SER) presents results of a review of the 340 Waste Handling Facility (340 Facility), Interim Safety Basis (ISB) (WHC-SD-WM-ISB-003, Rev 1). An extensive review was performed by Waste Programs Division (WPD), Quality, Safety and Health Division (QSH), and Waste Operations Division (WOD). The review was conducted to establish a technical basis for accepting risks associated with operations of the 340 Waste Handling Facility.

340 Waste Handling Facility is an existing facility located in the 300 Area of the Hanford Site. The current mission of the 340 Facility Complex is to provide quality receipt, storage, and disposal services in support of the Pacific Northwest National Laboratory (PNNL) low-level radioactive mixed waste discharges from five major buildings and other Hanford Site programs within the 300 Area.

The basis of the review included scope and content requirements specified in DOE-STD-3011-94, *Guidance for Preparation of DOE 5480.22 Technical Safety Requirements (TSR) and DOE 5480.23 Nuclear Safety Analysis Reports (SAR) Implementation Plans (IPs)*. In addition, the review followed the requirements in Waste Program Division (WPD) procedure 96-01, "Review and Approval of Safety Documentation," (WPD 1996), as well as requirements established by other DOE Orders, regulations, codes, standards, engineering, and scientific principles. A graded approach was used in developing a specific review criteria based on the risks the facility poses.

The ISB identifies a spectrum of potential hazards associated with the activities performed at the 340 Facility and evaluates consequences resulting from normal operations and potential accident scenarios. The ISB summarized the results of these evaluations and presented operational and institutional safety controls.

The ISB does not credit engineered safety features or operator/human intervention for protection of the general public or co-located workers during design basis events. The ISB has committed to ensure protection of the facility worker by implementation of procedures which comply with Institutional Safety Programs (ISP's) listed in Table 3-1 (e.g., Radiation Protection, ALARA, Occupational Safety, Fire Protection, Industrial Hygiene, Criticality Safety, Training for Nuclear Facilities, Radioactive Waste Management, Occurrence Reporting, Quality Assurance, Configuration Management, Conduct of Operations, Maintenance, Emergency Planning, and Environmental Protection). It is believed by complying with these ISP's, protection of the facility worker is adequately addressed.

The ISB concludes that the 340 Facility can be operated and maintained in compliance with applicable laws, regulations and departmental Orders and with acceptable risk to the health and safety of the facility workers, co-located workers, and members of the general public. It is concluded that the ISB

340 ISB SER

August, 1996

presents an adequate and acceptable basis for specific authorization of the analyzed operations at the 340 Facility.

TABLE OF CONTENTS

1.0	INTRODUCTION, OVERVIEW AND SUMMARY	1
2.0	INTERIM SAFETY BASIS REVIEW	4
2.1	Chapter 1 - Executive Summary	4
2.2	Chapter 2 - Introduction.	4
2.3	Chapter 3 - Design of Principle Structures, Components, Systems, Engineered Safety Features, and Processes	5
2.4	Chapter 3 - Accident Analysis	6
2.5	Chapter 3 - Safety Equipment Determination.	8
2.6	Chapter 3 - 340 Facility Administrative Requirements.	8
2.7	Chapters 4/5 - References Review	9
3.0	REFERENCES	10
	APPENDIX A: REVIEW TEAM MEMBERS	A-1

340 ISB SER

August, 1996

1.0 INTRODUCTION, OVERVIEW AND SUMMARY

Since 1989, the "Operating Limits for the 340 Facility, OSR-SW-152-00003, Rev A-0, the "Operating Specification Document for the 340 Waste Handling Facility", OSD-SW-153-0001, dated November 1990, and the "Safety Assessment Document for the 340 Waste Handling Facility", WHC-SD-WM-TM-001, Rev 0-A, dated 1994, provided the authorization basis for operations of the 340 Facility. In December, 1994, in response to the Safety Analysis Reports (SAR) and Technical Safety Requirements (TSR) Upgrade Implementation Plan (IP), WHC submitted the ISB, Rev 0, to DOE-RL for review and approval. The ISB was to be used for the purposes of maintaining an interim safety basis for operations.

DOE-RL established and conducted a structured review process, including technical review by the line organization (WPD), and a detailed review by the technical support organization for safety (QSH). Government Support contractors (GSSC) were used to provide technical expertise, as necessary. The comments generated through this process were formally resolved with WHC and DOE-RL (WPD & QSH). On June 1995, RL provided interim approval of the ISB (WHC-SD-WM-ISB-003).

Since June 1995, the 340 Facility has been using the 340 Facility Rev 0 ISB as the authorization basis.

RL's previous review of the ISB found that assumptions and conclusions were presented in the document without citing supporting reference documentation. The radiological inventory associated with the facility was partitioned for consideration in the hazard and accident analyses without supporting justification. WHC was requested to evaluate continued long term use of the 340 Facility and, based upon the results of this evaluation, determine the need for upgrading the existing SAR in accordance with DOE Order 5480.23. WHC's evaluation determined that the 340 Facility is sufficiently close to end of life, and thought it to be more cost effective to consider maintaining WHC-SD-WM-ISB-003 as the facility authorization basis in lieu of an upgraded SAR. The ISB was revised in December 1994 and WHC formally transmitted a revised ISB for RL review and approval.

In January 1996, an Occurrence Report initiated by RL was issued reporting a discovery of a potential USQ (WPD 1996) which revealed discrepancies between assumptions and controls associated with criticality protection in the approved authorization basis and its' reference document. WHC conducted an USQ evaluation in January 1996 to determine if in fact a potential facility segregation discrepancy existed between the 340 Facility solid waste storage exempt and isolated status, and between the criticality safety evaluation report and the ISB. It was determined that a USQ did not exist, however there were discrepancies found and corrected. In February 1996, WHC formally closed RL criticality assessment findings.

340 ISB SER

August, 1996

1.1 Overview of the WHC ISB

The purpose of the ISB, as described in Chapter 1.0 of WHC-SD-WM-ISB-003 (WHC 1996), is to "establish the Interim Safety Basis (ISB)..." and "...is a documented safety basis that provides the justification for the continued operation of a facility until an upgraded Final Safety Analysis Report (FSAR) is prepared...". Upon approval, the 340 Facility ISB, as supplemented by this SER, will be the Authorization Basis. The ISB was prepared under the WHC IP for DOE Orders 5480.22 and .23 (WHC 1992), but prior to issuance of DOE-STD-3011-94, *Guidance for Preparation of DOE 5480.22 Technical Safety Requirements (TSR) and DOE 5480.23 Nuclear Safety Analysis Reports (SAR) Implementation Plans (IPs)*. The Standard provides guidance on the content for Bases for Interim Operations (BIO) documentation. This ISB is, however, generally consistent with the guidance on-format and content contained in the Standard.

1.2 SER Review Process

The review of the ISB was performed in accordance with WPD 96-01, "Review and Approval of Safety Documentation" (WPD 1996). The process consisted of a phased review. The first phase of the review was conducted by representatives of WPD line management. The second phase of the review was conducted by a Peer Review Team, with appropriate technical backgrounds. The review team was comprised of representatives from DOE-RL's WPD, QSH and WOD. The evaluation included a review of many of the reference documents cited in the ISB, discussions with WHC personnel, and field observations of the facility to ensure a thorough understanding of the information and conclusions contained in the ISB.

Comments were collated and discussed by the Review Team for technical appropriateness. Comments were documented on Review Comment Record (RCR) forms which were subsequently transmitted to WHC for resolution. Proposed resolutions were developed by WHC personnel and forwarded to Review Team members. Team members individually reviewed the proposed resolution of their comments and further discussions between the Team members and WHC were conducted to resolve open issues. The RCR proposed responses were then revised by WHC to reflect the final dispositions.

The ISB was then revised incorporating the agreed upon resolutions and submitted to DOE-RL for approval in September 1996. The Review Team members reviewed the submittal and verified that it adequately addressed the agreed upon disposition of individual comments.

It should be noted that, in addition to the review of the ISB as discussed above, the review team is also crediting the technical review performed on Revision 0 of this ISB which was submitted in December 1994 (see Section 1.0 of this SER for discussion). Rev 0 of the ISB was approved by RL, but the approval was later found to be inadequate based

340 ISB SER

August, 1996

upon administrative considerations associated with identification of the approval authority.

The SER was prepared to guidance provided in WPD 96-01 (WPD 1996). The format is consistent with the content of recent SERs, DOE-HQ has found to be acceptable for other DOE facilities. The presentation follows the topics in the same order as presented in the submitted ISB (WHC-SD-WM-ISB-003, Rev 1) (WHC 1996). The ISB is not intended to be compliant with DOE Order 5480.23, but is written to demonstrate adequate level of safety basis for the 340 Facility operations. The 340 Facility is planned to initiate shutdown in the next two to five years, therefore a SAR for this short lived operations may not be warranted. As such, the ISB is a collection of existing safety documentation and newly prepared evaluations to support safe operations and serve as a basis for DOE 5480.21, Unreviewed Safety Question (USQ) evaluations.

The approach used to document the review and conclusions for this SER is to briefly summarize the safety basis information provided, evaluate the information provided against the requirements, and list significant factors important in making the decision of acceptability and approval of the safety basis. The SER then becomes a part of the overall safety basis, containing specific conditions and factors upon which approval of the ISB and operations at the 340 Facility is granted.

1.3 SER Conclusions

In conjunction with the imposition of the administrative control programs presented in the ISB, the ISB presents a consistent and adequate set of analyses to demonstrate the safety of current and planned operations at the facility. We conclude that the 340 Facility can be operated without unnecessary risk to the health and safety of facility workers, co-located workers, and undue risk to members of the general public. DOE-RL acceptance of this conclusion is predicated on WHC maintaining the Institutional Safety Programs described in Chapter 3.0 of the ISB for the protection of the facility worker.

The IOSR's in the ISB now address those portions of the former OSD-SW-153-00001 that still have relevance, (WHC 1990).

340 15B SER

August, 1996

2.0 INTERIM SAFETY BASIS REVIEW

2.1 Chapter 1 - Executive Summary Review

2.1.1 Description

Chapter 1.0 provides the purpose as well as defining the scope of the 15B. The 15B is based on revised facility and process systems which were discussed in the 340 Facility (DOE 1992c). The summary acknowledges since the 340 Facility is categorized as a Category 3 Nonreactor Nuclear Facility, only the facility worker, not onsite or offsite individual or environment, is at potential risk due to the operation of the facility. The 15B concludes the protection of the facility worker is assured by following WHC policies and facility procedures. A hazard analysis was performed which identified potential accidents at the facility. A set of bounding accidents were identified and analyzed. The results of the analysis confirmed the category of the facility.

2.1.2 Evaluation Findings

The information provided in Chapter 1.0 of the 15B is adequate to provide background and understanding of the purpose and scope of the 15B, definition of the facilities being evaluated, summary of the safety analyses performed in support of the 15B, and conclusions of safety of operations at the 340 Facility.

2.1.3 Conclusions

On the foregoing basis, the information presented in Chapter 1.0 is acceptable.

2.2 Chapter 2 - Introduction

2.2.1 Description

340 Facility design and operation, results of the hazard analysis are also presented and bounding accidents selected (a spray release, while filling the railroad tank car from one of the 340 Tank Vaults, and a solid waste fire). The bounding worst-case accidents determined from the PHA as having the highest severity probability and selected for further evaluation include: 1) a spray release while pumping liquid into a railroad tank car, and 2) a fire occurs in the 340-B West dry waste storage area. The analyses of these accidents presented showed that there are no unacceptable risks to the onsite worker or the public from operating the 340 Facility.

2.2.2 Evaluation Findings

Information presented in Chapter 2.0 is consistent with information in other sections of the ISB. In addition, the information adequately summarizes the hazards analyses presented in Chapter 3.0. All accident scenarios analyzed in Chapter 3.0 are summarized in Table 3-4 and Table 3-6, including dose consequences and frequency based guidelines.

2.2.3 Conclusions

On the foregoing basis, the information presented in Chapter 2.0 is acceptable. DOE-RL's acceptance of the risks associated with the accident scenarios is based on the analyses presented. DOE-RL's acceptance was not based on, or endorses, the WHC guidelines presented.

2.3 Chapter 3 - Design of Principal Structures, Components, Systems, Engineered Features, and Processes

2.3.1 Description

A description of principal structures, confinement systems, services and utilities, processes, and operations at 340 Facility was provided. The 340 Facility complex is described (Figure 1-1) and as covered by this evaluation includes the main 340-B East, 340-B East, 340 Vault Tanks, 340 Building, 340-A Above Ground Tanks, 3707-F RPS Basins Control Room, MO-336 Office Building, MO-741 RCT Trailer and the 307 Retention Process Sewer Basins.

The 307 Retention Process Sewer Basins was evaluated in 1995 and determined to be a non-nuclear facility separate from the overall 340 Facility. The 307 Basins, however, is within the boundaries of the 340 Facility fenced area.

Descriptions of the identification of codes and standards applicable to the design and construction of the facility is presented in conjunction with the description of the facility and operation.

Criticality Safety, Institutional Safety Programs, and Safety and Health Protection are also described in Chapter 3.0.

2.3.2 Evaluation Finding

The description of the 340 Facility and operations provided in ISB Chapter 3.0 was sufficiently complete, consistent with the hazard and accident analysis presented in the ISB, and consistent with respect to commitments to appropriate health and safety codes and standards. On that basis, Chapter 3.0 describes a facility and operations, with appropriate commitments to Institutional Safety Programs, that are adequate for protection of facility workers.

340 ISB SER

August, 1996

The 340 Facility is a Category 3 Nonreactor Nuclear facility, and hence, presents the potential for only significant localized consequences. To ensure the protection of the facility worker, 340 Facility has committed to Institutional Safety Programs. Appendix E, Table E-1, identifies the DOE Orders applicable to the various programs. Appendix E will become Appendix A when the ISB is revised to include RL comments. Based on a review of Appendix E, Table E-1, this approach is acceptable.

2.3.3 Conclusions

On the foregoing basis, the information presented in Chapter 3.0 is acceptable.

2.4 Chapter 3 - Hazard and Accident Analysis

2.4.1 Description

The 340 Facility identified DOE Order 5480.23 (DOE 1992b) and DOE STD-1027-92 (1992c) as the basis for facility hazard level categorization. The HA assumed unmitigated releases of radioactive material from the 340 Facility following bounding worst-case accident scenarios (a spray release in piping while pumping waste into a railroad tank car and a fire in the 340-B West dry-waste storage area). No credit was taken for engineered structures, systems, components, or human intervention.

2.4.2 Evaluation Findings

A Preliminary Hazards Analysis (PHA) was developed based on a walkdown of the 340 Facility including interviews of personnel and reviews of design and operations of the facility. Based on the facility being a Category 3 facility, this approach was considered acceptable.

Several assumptions were used in the development of the PHA. It is assumed that initiating events from one building will not create upset conditions in another building because of low potential energy release processes, and/or the fire suppression systems in these facilities.

Several facility worker scenarios were evaluated in the PHA, however, no severity levels were assigned to these scenarios. The ISB has committed to ensure protection of the facility worker by implementation of procedures which comply with Institutional Safety Programs (ISP's) listed in Chapter 3 (e.g., Radiation Protection, ALARA, Occupational Safety, Fire Protection, Industrial Hygiene, Criticality Safety, Training for Nuclear Facilities, Radioactive Waste Management, Occurrence Reporting, Quality Assurance, Configuration Management, Conduct of Operations, Maintenance, Emergency Planning, and Environmental Protection). It is believed by complying with these ISP's, protection of the facility worker is adequately addressed.

340 ISB SER

August, 1996

The basis for maximum radionuclides inventory was determined from data generated on the radioactive liquid waste and solid waste containing mixed fission products and other radionuclides, in the 340 Waste Handling Facility prior to its shipment for disposal. The ISB presented historical maximum concentrations revealed during years of sample analysis. These values were further increased (conservatively accounting for possible excursions over past observances) to function as bounding source terms. The isotopic break-out serves as the basis for substituting realistic, worst-case nuclides for alpha or beta/gamma totals (used because these gross parameter conform to the typical focus of both monitoring and laboratory screening analyses). Worst case depends upon the particular release scenarios and dose models used in the various evaluations.

A review of the 340 Facility data on Sr-90 and Pu-239 concentrations in samples were taken from the 340 Facility, 340 Vault Tanks, and the 340-B West Building solid waste between the years 1989 to 1993. This review showed that the highest concentrations of Sr-90 and Pu-239 in the samples analyzed for the 340 Vault Tank bounding source term in the facility inventory is 6.90 E-01 Ci beta/gamma and 1.16 E-03 Ci alpha and in the 15,000 gal tank is 3.93 E+04 Ci beta/gamma and 6.61 E+01 Ci alpha for hazard category calculations. The estimated total doses onsite and offsite is 3.92 rem effective dose equivalent (EDE) and 3.30 E-01 rem EDE, respectively.

Shipping record for the solid waste generated in the 340 Facility during the years 1989 through 1993 show that the maximum 208-L (55-gal) drum contained 3.08 E-03 Ci and the maximum plywood box contained 7.2 E-04 Ci. The higher curie levels for solid waste storage area values selected and used in the fire accident analysis are 10 Ci beta/gamma and 1.5 Ci alpha. The accident analysis in the ISB used these estimates of worst-case beta/gamma and alpha inventories for the onsite and offsite dose calculations. The total estimated dose onsite is 3.24 rem EDE and offsite is 2.72 E-01 rem EDE.

The ISB hazard analysis together with the accident analysis presented in concluded that unmitigated releases of radioactive material from the 340 Facility have the potential for significant localized consequences only.

The ISB concludes that all 340 Facility processes, and operations are enveloped within the Category 3 Hazard Classification defined in DOE 5480.23 (DOE 1992b). Based on a review of the radiological inventories and safety analysis presented in the ISB, this conclusion is acceptable.

It should be noted that RL requested additional information to be included in the ISB which identify uncertainties associated with the bounding accident's consequence and frequency. WHC agreed to insert the following into Section 3.1.5, P. 3-7 of the ISB: "The onsite risk for the solid waste fire (Figure 3-1), while relatively close to the OFFSITE guideline, is almost 10 time below the ONSITE guideline. The release

340. ISB SER

August, 1996

fraction uncertainties can be obtained from DOE/OHDBK-3010-94, P. 5-1. The ISB document used the bounding release fractions for the dose estimates. For a solid waste fire the bounding release fraction is 5 E-04 and median release fraction is 8 E-05 (P. 5-1). The expected onsite and offsite doses, using the median release fraction, is, respectively 0.518 rem and 0.0435 rem (verses the bounding values of 3.24 rem and 0.272 rem, respectively). The bounding values in the ISB are 6.25 times higher than the expected values and this difference represents the uncertainties for the consequences. The uncertainty for the frequency value, 1 E-03, is the range 1 E-04 to 1 E-02. The fire accident was judged to be an unlikely event with a frequency range of 1 E-04 to 1 E-02. 1 E-03 was the frequency value chosen for the figure to show the comparison to the guideline values."

2.4.3 Conclusions

The ISB adequately identified potential initiating events, energy sources and potential release paths. Hazardous chemical releases are expected to be less than quantities that require reporting under 40 CFR Part 302 during the postulated credible accident events. Chapter 2.0 and 3.0 provided sufficient and consistent site and facility information to support the hazard analysis. On the basis of the analyses of unmitigated releases, the classification of 340 Facility as a Hazard Category 3 non-reactor nuclear facility is acceptable.

The ISB also adequately identified the routine occupational hazards and DOE prescribed occupational safety and health standards that address these hazards. On the foregoing basis, the information provided in Chapter 3.0 is acceptable.

2.5 Chapter 3 - Safety Equipment Determination

2.5.1 Description

Chapter 3.0 focused on the need for Safety Class structures, systems, or components (SCSSC's) for preventative or mitigative functions.

2.5.2 Evaluation Findings

The ISB derived classification of structures, systems, and components using the definition of "Safety Class" identified in DOE 6430.1A, "General Design Criteria." The hazard analysis and accident analysis assumed unmitigated releases of radioactive material from facility operations. No credit was taken for engineered safety features or human intervention for protection of the general public and co-located workers. Therefore, the ISB did not identify any Safety Class structures, systems, and components (SCSSC).

2.5.3 Conclusions

340 ISB SER

August, 1996

On the foregoing basis, the information presented in ISB Chapter 3.0 is acceptable.

2.6 340 Facility Administrative Requirements

2.6.1 Description

The results of the accident analyses are presented in Table 3-4 (ISB). The results show there are no significant consequences to onsite workers or offsite public confirming the designation of the 340 Facility as a Category 3 Non-Reactor Nuclear Facility.

A limit to quantities of radioactive material which would produce accident dose consequences exceeding those in the bounding worst case accident scenario are proposed. The 340 Facility will implement an Inventory Control Program. An Administrative Requirement is established in ISB Chapter 3.0 to implement the Inventory Control Program.

No credit was taken in the accident analysis for Safety Class SSC's. In addition, no Safety Limits (SL), Limiting Conditions for Operations (LCO), or Limiting Control Settings (LCS's) have been identified.

2.6.2 Evaluation Findings

No credit was taken in the accident analysis for Safety Class SSC's, therefore, the only control necessary to protect public and the on-site worker is the interim operational safety requirements (IOSR) and facility liquid and solid waste radiological inventory.

The OSL's presented in the "Operating Limits for the 340 Facility, OSR-SW-152-00003, Rev A-0, the "Operating Specification Document for the 340 Waste Handling Facility, OSD-SW-153-0001, dated 1990, and the "Safety Assessment Document for the 340 Waste Handling Facility", WHC-SD-WM-TM-001, Rev 0-A, dated 1994, are no longer required to be maintained. The OSL's were established to ensure specific equipment or systems, deemed important to safety, were operational in nature. Since no credit is taken in the accident analysis for engineered safety features, deletion of the OSLs is acceptable.

Commitments for the protection of the facility workers is assured in ISB Chapter 3.0 through Administrative Controls/Operational Restrictions and Institutional Safety Programs.

2.6.3 Conclusions

On the foregoing basis, the information presented in Chapter 3.0 is acceptable.

340 ISB SER

August, 1996

2.7 References

2.7.1 Description

The documents cited in the ISB are listed in Appendix E, Table E-1. These references include Federal and State laws, WHC manuals, evaluations and reports, and DOE Orders. Applicable federal regulations and laws, controlled manuals, and other pertinent safety documentation are listed in Chapter 4 and Chapter 5. These cited references provide much of the basis information upon which the safety analyses and risk assessments are performed, as well as the governing rules, regulations and criteria for ensuring operations at the 340 Facility are safely performed.

2.7.2 Evaluation Findings

The references provided appear to be comprehensive and sufficient to identify pertinent information necessary to ensure that operations at the 340 Facility will continue to be conducted in a safe and environmentally acceptable manner with appropriate regulatory controls.

2.7.3 Conclusions

The references cited are adequate for providing necessary information on the safety bases and regulatory requirements in place at the 340 Facility.

340 ISB SER

August, 1996

3.0 REFERENCES

- DOE 1991 "Unreviewed Safety Questions", DOE Order 5480.21, U.S. Department of Energy.
- DOE 1992a "Technical Safety Requirements", DOE Order 5480.22, U.S. Department of Energy.
- DOE 1992b "Nuclear Safety Analysis Reports", DOE Order 5480.23, U.S. Department of Energy.
- DOE 1992c "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order-5480.23, Nuclear Safety Analysis Reports", DOE Standard DOE-STD-1027-92, U.S. Department of Energy, December, 1992.
- DOE 1994a "Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports", DOE Standard DOE-STD-3009-94, U.S. Department of Energy, July, 1994.
- DOE 1994b "Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans", DOE Standard DOE-STD-3011-94, U.S. Department of Energy, November, 1994.
- WHC 1990 "Operating Specification Document for the 340 Waste Handling Facility", OSD-SW-153-0001, Westinghouse Hanford Company, Richland, WA., dated November 7, 1990.
- WHC 1992 "Implementation Plan for DOE Orders 5480.21, 5480.22 and 5480.23", External Letter 9257875 from J. M. Knoll, Jr., Westinghouse Hanford Company to R. D. Larson, Richland Operations Office, dated October 28, 1992, and Rev 1, dated August 20, 1993.
- WHC 1995 "Safety Analysis Manual", WHC-CM-4-46, Westinghouse Hanford Company, Richland, WA.
- WHC 1996 "340 Waste Handling Facility Interim Safety Basis", WHC-SD-WM-ISB-003, Rev 1, Westinghouse Hanford Company, Richland, WA.
- WOD 1996 "Criticality Protection Program Assessment Report", A-95-WOD-300LEF-001, DOE/RL Waste Operations Division, Richland, WA.
- WPD 1996 "Review and Approval of Safety Documentation", WPD 96-01, DOE/RL Waste Programs Division, Richland, WA.

340 15B SER

August, 1996

APPENDIX A: REVIEW TEAM MEMBERS

340 Facility 15B

Anna V. Beard, DOE/RL WPD
(Lead Program Engineer)

Review Team Members:

Joe Voice, DOE/RL WOD
(340 Facility Representative)

Stacy Helmann, DOE/RL WOD
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(Nuclear Safety)

Tim Powell, MACTEC
(Independent Safety)

RUST

FEDERAL SERVICES OF HANFORD

Manual Document Change**MDC INITIATOR: COMPLETE BLOCKS 1-7**

1. MANUAL NUMBER	2. MANUAL TITLE		
WHC-SD-WM-0003	340 Waste Handling Facility Interim Safety Basis		
3. SECTION NUMBER	4. SECTION TITLE	5. REV. NO. (PER MANUAL VERSION)	
		1	
6. ORGANIZATION	INITIATOR	MSIN	L6-05
300 Area Liquid Effluent Facilities	R. T. Stordeur	PHONE	372-3452
		TODAY'S DATE	10/01/96

7. PROPOSED CHANGE

This manual/procedure was found acceptable for incorporation and implementation by Rust Federal Services Hanford (RFSH) with the following changes:

- 1) All references to Westinghouse Hanford Co. (WHC) shall be considered as RFSH or in the broader sense Fluor Daniel Hanford (FDH) or the applicable FDH team company.
- 2) All references to WHC level II or level III manuals shall be considered as references to the applicable FDH team replacement manual.

These changes will be incorporated into this manual during the next planned review/revision which must occur not later than October 1, 1997.

RESPONSIBLE ORGANIZATION: COMPLETE BLOCKS 8-10**8. RESPONSIBLE ORGANIZATION DISPOSITION**

300 Area Liquid Effluent Facilities

9. DISPOSITIONED BY

L. W. Roberts

*LW Roberts***10. DATE**

10/1/96

MEETING MINUTES				
SUBJECT: <u>HASP Requirements</u>				
TO: <u>L. W. Roberts</u>		BUILDING: <u>300 LEF</u>		
FROM: <u>M. W. CLAYTON</u>		CHAIRMAN: <u>L. W. Roberts</u>		
DEPARTMENT-OPERATION-COMPONENT	AREA	SHIFT	DATE OF MEETING	NUMBER ATTENDING
<u>RFSH</u>	<u>300</u>	<u>—</u>	<u>3/12/97</u>	<u>4</u>

- Met to discuss Requirements for Health and Safety Plans (HASPs) at 310 and 340.
 - o Reviewed the Requirements of 29CFR1910.130 HNF-CH-4-40 Section 2.19 + 2.20 and DOE-EM-STD-5502-94 (Aug 1994)
- Consensus of attendees was that there was no regulatory driver for development of a HASP at either facility. The 340 ISB and facility operating and administrative procedures incorporate appropriate worker safety requirements. 310 SAD, administrative and operating procedures incorporate appropriate worker safety and health requirements.
 - o No further actions required

MEETING ATTENDANCE ROSTER

NAME	ORGANIZATION
<u>M. W. CLAYTON</u>	<u>RFSH</u>
<u>DEB WILLIAMS</u>	<u>DOE</u>
<u>L. W. Roberts</u>	<u>RFSH</u>
<u>ANNA V. BEARD</u>	<u>DOE</u>