Plutonium Finishing Plant Safety Evaluation Report

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**U.S. Department of Energy
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EXECUTIVE SUMMARY

The Plutonium Finishing Plant (PFP) is operated by Westinghouse Hanford Company (WHC) for the U.S. Department of Energy, Richland Operations Office (DOE/RL). The PFP is located on the 570 mi² Hanford Site in south central Washington state. The facility, classified as a Hazard Category 2 facility, has operated since 1949 performing various defense related missions involving the processing and handling of plutonium. Since its construction, there have been several major additions to the facility and many upgrades to enhance safety. Westinghouse Hanford Company reviewed the facility's 45 years of operating experience and found that no significant radiological or toxicological accidents had occurred that resulted in adverse health effects to the public.

The PFP Final Safety Analysis Report (FSAR) was prepared by WHC to document the hazards associated with the facility, present safety analyses of potential accident scenarios, and demonstrate the adequacy of safety class structures, systems, and components (SSCs) and operational safety requirements (OSRs) necessary to eliminate, control, or mitigate the identified hazards. The FSAR establishes the safety basis upon which DOE depends for its conclusion that activities at the facility can be conducted safely.

The PFP FSAR was originally prepared between 1988 and 1990 in accordance with DOE Order 5481.1B, *Safety Analysis and Review System*, prior to issuance of DOE Order 5480.23, *Nuclear Safety Analysis Reports*. The PFP FSAR's content and format are based on guidance from U.S. Nuclear Regulatory Guide (RG) 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*, as prescribed by DOE Order 5480.5. A draft FSAR was submitted to DOE for review and approval in January 1991. In October 1992 WHC submitted to DOE/RL an implementation plan for complying with DOE Orders 5480.21, 5480.22, and 5480.23. The implementation plan indicated that because of the absence of a long term operating mission for PFP, and the currently envisioned shutdown of the facility, the PFP FSAR will not be upgraded to comply with DOE Order 5480.23. The implementation plan was forwarded to DOE's Office of Environmental Management (EM) in August 1993, and in April 1994 the plan was approved by DOE/RL. Westinghouse Hanford Company management has agreed to maintain the current FSAR by performing annual updates and by implementing the Unreviewed Safety Question (USQ) process.

When the FSAR was submitted in January 1991, it was envisioned that the facility would process stored plutonium nitrate solutions and unstable solids into a chemically stable form of plutonium oxide (referred to as the "stabilization run"). Westinghouse Hanford Company also planned to begin preparing the facility for decontamination and decommissioning. In 1993, public meetings were held to present the planned activities. As a result of discussions with the public and stakeholder groups, it was decided that an Environmental Impact Statement (EIS) should be prepared to document the environmental impact associated with the planned stabilization run. An EIS is currently being developed to address the longer term use options

such as decontamination and decommissioning and other projected options for the facility. Until the EIS is issued, the facility's primary operational objectives other than plutonium storage and related activities are not well defined. Westinghouse Hanford Company has completed an Environmental Assessment (EA) for stabilizing scrap material stored in processing gloveboxes at PFP.

SCOPE OF REVIEW

The PFP FSAR review and evaluation effort was conducted by the U.S. Department of Energy (DOE), Office of Defense Programs (DP), Office of Engineering, Operations, Security & Transition Support (DP-31) in support of what is now the Office of SRS Facilities (DP-33). Technical assistance was provided by safety analysts at Pacific Northwest Laboratory (PNL) and Idaho National Engineering Laboratory (INEL). In June 1992, a detailed review of WHC's accident safety analyses, OSRs, and SSCs was initiated. Evaluation criteria for the review were primarily derived from requirements established in DOE Order 5481.1B, RG 3.39, and WHC guidelines for assessing risk and identifying OSRs and SSCs. A list of 150 technical questions requesting clarification and/or justification of specific information presented in the PFP FSAR was prepared in July 1992 and transmitted to WHC by DOE. Westinghouse Hanford Company responded to the initial list of questions and by November 1992 all but nine technical issues had been resolved. The initial review comments and WHC responses are included as part of this SER in Appendix A.

To resolve the remaining issues, meetings were arranged between DOE, WHC, PNL, and INEL staff members. In March 1993, an agreement was reached with WHC on a strategy to resolve the remaining nine open issues. Due to the complexity of the required re-analyses and additional technical reviews, the open issues were not completely resolved until May 1994. Between May and December 1994 revisions were made to the Accident Safety Analyses Chapter 9.0 as required to incorporate the resolution of the nine open issues. The following are the most significant changes made to the PFP FSAR as a result of the review process:

- significant upgrades were made to the facility's Operational Safety Requirements (OSRs) to adequately describe the operating limits for the facility;
- the facility's list of Safety Class SSCs was updated;
- a program was established to control the facility's plutonium inventory to ensure that facility operations are conducted within the bounds of the safety analyses;
- a re-analysis of glovebox MT-5 operations demonstrated that the risks associated with processes in the glovebox are acceptably low;
- a re-analysis of flammable gas used within the facility identified the need for additional administrative controls to be included in the facility's OSRs;

- the seismic event was re-analyzed assuming the maximum allowable plutonium inventory levels;
- recommended seismic upgrades that were identified in the 1991 draft FSAR, except for filterboxes 7A and 9A, are completed and documented as such in the PFP FSAR (1994) (filterboxes 7A and 9A are assumed to fail in the current seismic analysis);
- an additional accident scenario involving waste drums stored outside the facility was analyzed and incorporated into the safety analysis chapter;
- a re-analysis of both the filtrate evaporator and product concentrator revealed the necessity to install additional passive engineered safety features prior to operation. Due to the uncertain mission pending outcome of the EIS, WHC decided not to pursue authorization to operate this equipment. Thus, the accident analyses for this equipment were removed from the PFP FSAR;
- a more conservative method for determining the source term was developed for use in the various accidents analyzed in the safety analysis chapter;
- a seismic analyses effort was undertaken by WHC to provide documentation to support the seismic qualification of the 2736 complex.

Documented in this Safety Evaluation Report (SER) is DOE's independent review and evaluation of the FSAR and basis for approval of the PFP FSAR. The evaluation is presented in a format that parallels the format of the PFP FSAR.

LIMITING ACCIDENT SCENARIO - SEISMIC EVENT

The most limiting accident scenario identified at PFP in terms of the most significant radiological dose consequences was determined to be a seismic event. Westinghouse Hanford Company performed a seismic assessment of the facility to determine how the individual gloveboxes, fume hoods, and components would respond to an earthquake. The assessment describes the postulated room-by-room damage that may occur as a result of a 0.20 g Evaluation Basis Earthquake (EBE), which is referred to in the FSAR as a Design Basis Earthquake (DBE). For consistency the EBE is also referred to as a DBE in this SER. The potential for seismically induced criticalities and fires was also documented in the FSAR. To determine a bounding (i.e. worst case) seismic scenario WHC assumed that the following processes were operating: the RMC line; PRF solvent extraction processes; PRF miscellaneous treatment processes; Product handling; Laboratory operations conducted in the Analytical, Plutonium Process Support, and Standards Laboratories; and Vault storage activities.

Westinghouse Hanford Company demonstrated that the probability of significant structural damage to buildings 234-5Z, 236-Z, 242-Z, 291-Z, 2736-Z, ZA, and ZB, and the 291-Z stack

is less 1×10^{-4} /yr. This annual probability of structural damage is based on a vibratory ground motion of 0.20 g peak ground acceleration (pga) event as the EBE. The 232-Z Building, constructed of reinforced concrete block wall panels, was used in the early 1970s to incinerate plutonium contaminated waste. This building was not shown to survive a 0.2 g seismic event. The postulated release from this building contributes approximately 40% to the total radiological dose from the seismic event which WHC has shown to be within risk acceptance guidelines. It is shown that the 232-Z Building will survive a 0.09 g pga seismic event which is associated with an occurrence frequency of 2×10^{-3} /yr.

Review of WHC's initial assumptions regarding the seismic qualification of various equipment located within the 2736 complex revealed that inadequate seismic analyses existed to support the seismic qualification of the equipment. The equipment has been classified as Safety Class 1. Westinghouse Hanford Company has committed to perform dynamic analysis of the equipment and perform any necessary seismic upgrades which might be required to qualify the equipment (Ruff 1994). Since this discrepancy involves a condition in an existing facility that is currently operating, and since resolution of this discrepancy is being aggressively pursued by WHC, DOE has agreed to the schedule of completion as outlined in Ruff (1994).

Westinghouse Hanford Company developed two separate release scenarios for the seismic event. The first scenario assumes an earthquake occurs disrupting electrical power to the facility's main ventilation supply and exhaust fans and also preventing the backup steam driven fans from operating. For this case, it is postulated that wind blowing through open doors and building cracks, damage resulting from the earthquake, will result in a ground level point release. This scenario is referred to as the "No-Vent" case and results in the most limiting radiological dose to the co-located worker, or onsite receptor. The second scenario assumes that the earthquake does not disrupt the electrical power supplied to the fans, however, the supply ventilation fans are shutdown by an installed seismic shutdown switch. For this case, plutonium is transported out of the facility through the 291-Z stack via the ventilation exhaust system (failure of two stages of HEPA filters is assumed). This scenario is referred to as the "Vent" case and results in the maximum radiological dose to the public, or offsite receptor.

Both cases relating to the seismic event were reviewed in detail. The review determined that sufficient detail and justification were provided to ascertain that the predicted level of damage was appropriate. For example, if sufficient evidence existed to demonstrate the seismic qualification of a glovebox, fume hood, component and/or system then credit was taken by WHC for its seismic integrity and no release was postulated. If insufficient evidence existed to demonstrate its seismic qualification then a release was postulated and calculated. The amount of material available for release from the facility was conservatively assumed to be dispersible plutonium equal to all of the gloveboxes simultaneously loaded to the maximum permitted by the OSRs, which for most gloveboxes equals the Criticality Prevention Specification (CPS) limits.

The focus of the review was to ensure that the assumptions made by WHC were technically justified and that the numerical values used to calculate the radiological dose were adequately conservative and bounding. The review evaluated the resultant radiological dose calculations based on the postulated release caused by a seismic event and determined that the calculations were acceptable. Based on the assumptions used in the analyses, the plutonium inventory assumed in the analysis, and the calculational methods employed, the review concluded that seismic analyses were sufficiently conservative and provide bounding radiological dose consequences. The risk associated with a 0.2 g seismic event was shown to be acceptable based on WHC's risk acceptance guidelines.

SELECTION CRITERIA FOR SSCs AND OSRs

Westinghouse Hanford Company established criteria for defining Safety Class SSCs. Safety Class 1 and 2 SSCs were identified as 1) structural confinement boundaries necessary to prevent and/or mitigate the release of radioactive material to the environment, 2) ventilation confinement boundaries necessary to prevent and/or mitigate the release of radioactive material to the environment, 3) equipment that prevents explosions, prevents damage to structural or ventilation confinement systems, or is used to maintain or monitor the integrity of the ventilation confinement boundary, 4) equipment installed to provide an essential barrier utilized in the double contingency principle for prevention of nuclear criticality, and 5) alarm systems used to evacuate personnel in the event of a nuclear criticality. Safety Class 3 SSCs were identified as 1) internal confinement features (i.e. facility ventilation zones) that prevent the spread of contamination within the facility, 2) systems related to fire protection, 3) equipment used to warn personnel of high airborne contamination in facility working areas, and 4) equipment necessary for the safe evacuation of facility personnel in the event of an emergency.

In general, Safety Class 1 and 2 SSCs are relied upon to prevent or mitigate adverse health effects to the public and co-located worker in the event of an accident. Safety Class 3 SSCs protect facility workers by ensuring the confinement of hazardous materials and notification of abnormal conditions which warrant evacuation from the facility. Safety Class 3 SSCs are subject to standard design, fabrication, and industrial test standards and codes.

Westinghouse Hanford Company developed OSRs for the facility based on defining the facility's operating conditions, safe boundaries, and management and administrative controls necessary to prevent or mitigate the adverse consequences of an accident on the health of the public, co-located worker, or facility workers. Specifically, selection criteria were based on numerical threshold radiological and toxicological dose criteria in FSAR Section 9.2.4 and DOE Order 5480.22 and its predecessors.

CONCLUSIONS

The PFP FSAR was prepared in accordance with DOE Order 5481.1B, since it was initiated prior to the April 30, 1992 effective date of DOE Order 5480.23. The format and content are based on U.S. Nuclear Regulatory Guide (RG) 3.39 which DOE considers appropriate for this application. The DOE review concludes that the PFP FSAR meets the requirements of DOE Order 5481.1B and RG 3.39. Because of PFP's age and limited future functional requirements the PFP FSAR will not be revised to comply with DOE Order 5480.23. Westinghouse Hanford Company management will maintain the current FSAR by performing annual updates and properly implementing the USQ process.

Because of ongoing facility modifications, the time required to prepare and review an FSAR, changing facility missions, and changing DOE safety requirements, various information throughout chapters 1-8, 10, 11, and 12 is currently out-of-date. Where appropriate, the DOE review sought out current information and documented the discrepancies regarding out of date information in this SER. For example, WHC's Environmental Compliance Manual (ECM) provides the most up-to-date information regarding the monitoring of effluent waste streams as opposed to the out-dated requirements presented in Chapter 7.0. Westinghouse Hanford Company has committed to revise out-dated material during the first annual update to the PFP FSAR. Chapter 11 has been replaced by OSRs documented in WHC-SD-CP-OSR-010 which have been approved by DOE.

The DOE review concludes that sufficient conservatism exists and an appropriate level of detail is provided to assess the risks associated with the facility. Furthermore, the safety analyses prepared for each identified accident scenario were reviewed in sufficient detail to conclude that the radiological doses and toxicological dosages calculated in the PFP FSAR provide conservative bounding analyses of credible accident scenarios. The radiological risks identified at the facility exceed the toxicological risks. A comparison of the radiological risks identified at PFP to WHC's risk acceptance guidelines for the onsite worker and the public are shown in Figures 1 and 2 respectively.

Figures 1 and 2 display WHC's radiological risk acceptance guidelines. Each data point plotted in Figures 1 and 2 correspond to a postulated accident analyzed in the PFP FSAR. The probabilities and doses plotted are based on a number of conservatisms applied throughout the safety analyses and rely on the OSRs and SSCs established in the PFP FSAR. The figures provide a graphical representation of the facility's "risk envelope" and demonstrate that the identified bounding accidents meet WHC risk acceptance guidelines which are well below the DOE safety goals in SEN-35-91, *Nuclear Safety Policy*.

In conclusion, the PFP FSAR properly documents the hazards associated with the facility's operations, adequately defines the safety class SSCs necessary for plant safety, and establishes the minimum required OSRs for safe plant operations. A program has been established to control the facility's plutonium inventory to ensure that facility operations are conducted within the bounds of the defined safety envelope. Furthermore, the DOE review concludes

that the risks associated with conducting plutonium handling, processing, and storage operations within PFP facilities, as described in the PFP FSAR, are acceptable, since the accident safety analyses associated with these activities meet WHC risk acceptance guidelines and DOE safety goals in SEN-35-91. Based on WHC's commitment to complete the seismic evaluation of equipment located within the 2736-Z complex, formal acceptance and approval of the PFP FSAR (1994) is granted as of the issuance of this report.

Figure 1. A comparison of PFP's onsite risks to WHC's risk acceptance Guidelines for the onsite receptor.

Figure 2. A comparison of PFP's offsite risks to WHC's risk acceptance guidelines for the offsite receptor.

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ACRONYMS LIST

AC	Administrative Controls
ARF	Airborne Release Fraction
ASME	American Society of Mechanical Engineers
CA	Plutonium Extraction Column
CAM	Continuous Air Monitor
CAS	Criticality Alarm System
CC	Pulsed Stripping Column
CCX	Solvent Extraction Column
CPS	Criticality Prevention Specifications
CSA	Compliance Schedule Agreement
CSE	Criticality Safety Evaluations
DBE	Design Basis Event
DCG	Derived Concentration Guide
DF	Design Features
DOE	Department of Energy
DOE/RL	Department of Energy Richland Operations Office
DOT	Department of Transportation
DP	Defense Programs
ECM	Environmental Compliance Manual
EDE	Effective Dose Equivalent
EIS	Environmental Impact Statement
EM	Environmental Management
EPA	Environmental Protection Agency
ERPG	Emergency Response Planning Guide
ERO	Emergency Response Organization
ESFs	Engineered Safety Features
FAA	Federal Aviation Administration
FSAR	Final Safety Analysis Report
GOCO	Government Owned Contractor Operated
HEDOP	Hanford Environmental Dose Overview Panel
HEPA	High Efficiency Particulate Air (filters)
HF	hydrogen fluoride
HFD	Hanford Fire Department
ICRP	International Commission on Radiological Protection
IDLH	Immediately Dangerous to Life and Health
INEL	Idaho National Engineering Laboratory
LCF	Limited Control Facility
LCO	Limiting Condition for Operation
LCS	Limiting Control Setting
LANL	Los Alamos National Laboratory
LOI	loss-on-ignition

LPG	Liquid Propane Gas
MCW	Maximally Exposed Co-Located Worker
MI	Maximally Exposed Individual
MOI	Maximally Exposed Offsite Individual
NCRP	National Council on Radiation Protection and Measurements
NDA	Nondestructive Analysis
NESHAPS	National Emission Standards for Hazardous Air Pollutant
NRC	Nuclear Regulatory Commission
NTSB	National Transportation Safety Board
OJT	On the Job Training
OSHA	Occupational Safety and Health Administration
OSD	Operating Specification Document
OSR	Operational Safety Requirements
PFP	Plutonium Finishing Plant
PNL	Pacific Northwest Laboratory
PR	Product Receiver
PRF	Plutonium Reclamation Facility
PRT	Prereduction Tank
QA	Quality Assurance
RAG	Risk Acceptance Guidelines
RF	Respirable Fraction
RG	Regulatory Guide
RMA	Remote Mechanical A Pressure Line
RMC	Remote Mechanical C Pressure Line
SAR	Safety Analysis Report
SE	Safety Envelope
SEAC	Safety Evaluation Advisory Council
SER	Safety Evaluation Report
SL	Safety Limit
SME	subject matter experts
SNM	Special Nuclear Material
SSC	Structures, Systems, and Components
SSI	Soil Structure Interaction
TBP	tri-butyl phosphate
TRU	transuranic
USNRC	United States Nuclear Regulatory Commission
USQ	Unreviewed Safety Questions
WHC	Westinghouse Hanford Company

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1.0 INTRODUCTION

The Plutonium Finishing Plant (PFP) is operated by Westinghouse Hanford Company (WHC) for the U.S. Department of Energy, Richland Operations Office (DOE/RL). The PFP, previously known as the Plutonium Process and Storage Facility, or Z-Plant, was built and put into operation in 1949. The PFP provided the Hanford site with the capability to purify concentrated plutonium nitrate solution, reduce the solution to plutonium metal, and fabricate the metal into useful parts. Since 1949 PFP has been used for various processing missions, including plutonium purification, oxide production, metal production, parts fabrication, plutonium recovery, and the recovery of americium (Am-241). The PFP has also been used for receipt and large scale storage of plutonium scrap and product materials. The PFP includes the following buildings or facilities:

- 234-5Z Building, Plutonium Finishing Plant (PFP)
- 236-Z Building, Plutonium Reclamation Facility (PRF)
- 242-Z Building, Waste Treatment Facility
- 241-Z Building, Waste Solution Facility
- 291-Z Building, Exhaust Fan House
- 2736 Complex, Plutonium Storage Facility (PSF)
- and various associated support buildings.

The PFP is located on the 570 mi² Hanford Site in the southeast corner of Washington state. The site, a semiarid upland plateau in the Pasco Basin, is bordered by a series of anticlinal ridges and the Columbia River. The Hanford Site boundary delineates the area considered as an exclusion area per 10 CFR 100 and access to the Hanford Site is controlled by DOE. There are no residents within the Hanford Site boundary. Within this boundary, only DOE, contractor personnel, or other authorized persons are allowed to travel. The Hanford Site is traversed by State Highway 240, 2.5 mi to the west of PFP and the Columbia River approximately 7 mi to the north. The closest resident to the PFP is approximately 8 mi due west. This natural isolation reduces any potential offsite impact due to abnormal events or potential accidents.

The PFP Final Safety Analysis Report (FSAR) was prepared (1988-90) by WHC to document the hazards associated with the facility, present safety analyses of potential accident scenarios, and demonstrate the adequacy of safety class structures, systems, and components (SSCs) and operational safety requirements (OSRs) necessary to eliminate, control, or mitigate the identified hazards. Westinghouse Hanford Company prepared the PFP FSAR in accordance with DOE Order 5481.1B, *Safety Analysis and Review System*, prior to issuance of DOE Order 5480.23, *Nuclear Safety Analysis Reports*. The PFP FSAR's content and format are based on guidance from U.S. Nuclear Regulatory Guide (RG) 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*, as prescribed by DOE Order 5480.5. Facility operations and activities addressed in the PFP FSAR include, but are not limited to, the following:

- Engineering Development Lab
- Process Support Laboratory
- RMC Line Calciner process for oxide conversion
- Pu Handling and Storage Operations
- MT-5 Glovebox Operations
- Ventilation Systems
- 2736-Z Pu Storage Complex Receiving, Sorting, Storing, and Shipping Operations
- Maintenance shops
- Instrument shops
- 241-Z Waste Solution Facility
- Waste drum storage outside of the facility

Equipment operation and processes specifically not analyzed in the PFP FSAR and therefore not authorized by this SER are listed below.

- Pu metal production
- RMC Line, Pu metal production, and use of pressure vessels
- Product Concentrator operations
- Filtrate Evaporator operations
- Laboratory Waste Concentrator
- Fluorinator
- Use of Hydrogen Fluoride (HF) gas
- Use of Acetylene gas
- Slag and Crucible System Deentrainer
- Thermal stabilization (metal): 234-5Z/HA-21I
- Electrolytic metal dissolution: 236-Z/MT-3
- Incinerator ash calcination: 234-5Z/HA-40F
- Incinerator ash dissolution: 234-5Z/H6-46F
- Polycube calcination: 236-Z/MT-4
- RMA Line metal and oxide production gloveboxes
- Remediation piping cleanup: 234-5Z/235B-1
- operation of 232-Z incinerator: 232-Z

Future operation of these processes and/or equipment will require additional safety analyses to be prepared.

A draft FSAR was submitted to DOE for review and approval in January 1991. In October 1992 WHC submitted to DOE/RL an implementation plan for complying with DOE Orders 5480.21, 5480.22, and 5480.23. The implementation plan indicated that because of the absence of a long term operating mission for PFP, and the currently envisioned shutdown of the facility, the PFP FSAR will not be upgraded to comply with DOE Order 5480.23. The implementation plan was forwarded to DOE's Office of Environmental Management (EM) in August 1993, and in April 1994 the plan was approved by DOE/RL. Westinghouse Hanford

Company management has agreed to maintain the current FSAR by performing annual updates and by implementing the Unreviewed Safety Question (USQ) process.

The FSAR establishes the safety basis upon which DOE depends for its conclusion that activities at the facility can be conducted safely. When the draft FSAR was submitted in January 1991, the facility planned to start up and process stored plutonium nitrate solutions and unstable solids into a chemically stable form of plutonium oxide (this plan was referred to as the "stabilization run"). Westinghouse Hanford Company also planned to prepare the facility for decontamination and decommissioning activities. In 1993, public meetings were held to present the planned activities. As a result of discussions with the public and stakeholder groups, it was decided that an Environmental Impact Statement (EIS) should be prepared to document the environmental impact associated with the planned stabilization run. Westinghouse Hanford Company recently prepared an Environmental Assessment (EA) for stabilizing the scrap material stored in processing gloveboxes at PFP. An EIS is currently being developed to address the long term use options such as decontamination and decommissioning and other projected options for the facility. Until the EIS is issued, the facility's primary operational objectives other than Pu storage and related activities are not well defined.

The PFP FSAR review and evaluation effort was conducted by the U.S. Department of Energy (DOE), Office of Defense Programs (DP), Office of Engineering, Operations, Security & Transition Support (DP-31) in support of what is now the Office of SRS Facilities (DP-33). Technical assistance was provided by safety analysts at Pacific Northwest Laboratory (PNL) and Idaho National Engineering Laboratory (INEL). In June 1992, a detailed review of WHC's accident safety analyses, OSRs, and SSCs was initiated. A list of 150 technical questions requesting clarification and/or justification of specific information presented in the PFP FSAR was prepared in July 1992 and transmitted to WHC by DP-31. Westinghouse Hanford Company responded to the initial list of questions and by November 1992 all but nine technical issues had been resolved. The initial review comments together with the WHC responses are included as part of this report in Appendix A.

To resolve the remaining issues, meetings were arranged between DOE, WHC, PNL, and INEL staff members. In March 1993, an agreement was reached with WHC on how to resolve the remaining nine open issues. Due to the complexity of the required re-analyses and the additional technical reviews, the open issues were not completely resolved until May 1994. Revisions were made to the Accident Safety Analyses Chapter 9.0 as required to incorporate resolution of the nine open issues.

Documented in this Safety Evaluation Report (SER) is DOE's independent review and evaluation of the PFP FSAR and the basis for approval of the PFP FSAR. The evaluation is presented in a format that parallels the format of the PFP FSAR. As an aid to the reader, a list of acronyms has been included at the beginning of this report.

A graded approach was used to review the PFP FSAR. As a result, chapters 1, 2, 3, 5 and 6 were not specifically evaluated in this SER. These chapters provide general engineering details and data relating to site characteristics, plant design, and plant processes and were written in accordance with guidelines in RG 3.39. The information in these chapters was reviewed as necessary to support the review and evaluation of the FSAR. The accident safety analyses presented in Chapter 9 along with the OSRs presented in WHC-SD-CP-OSR-010 were reviewed in greater detail. The OSRs presented in Chapter 11 of the PFP FSAR have been replaced by OSRs established in WHC-SD-CP-OSR-010 which was approved by DOE in Donald F. Knuth's May 20, 1994 memorandum to J. Wagoner titled, "Approval of Plutonium Finishing Plant (PFP) Operational Safety Requirements (OSRs)."

In 1991, DOE issued Secretary of Energy Notice SEN-35-91, "Nuclear Safety Policy" establishing a foundation for developing risk based guidelines for accidental radiological releases. Westinghouse Hanford Company developed risk acceptance guidelines (RAGs) based on the concept that radiological and toxicological consequences associated with a higher probability accident should be less severe than radiological and toxicological consequences associated with a lower probability accident. Westinghouse Hanford Company's risk acceptance guidelines are more conservative than SEN-35-91 and are based on various regulatory limits which were originally developed to control routine exposures and provide siting criteria for commercial nuclear reactors.

A summary of the major accidents, natural phenomena events, and external events analyzed in the PFP FSAR is provided in Table 1. For each analyzed accident the associated radiological and/or toxicological dose consequences, estimated event frequency, and a comparison to WHC's risk acceptance guidelines are presented. Accidents having an estimated annual frequency of occurrence less than $10^{-6}/\text{yr}$ are considered incredible, and therefore, related consequence analyses were not performed. However, in the case of the Remote Mechanical C (RMC) line fire, WHC developed a deterministic analysis to identify the risks associated with a worst-case bounding fire scenario. The DOE review effort concluded that the PFP FSAR properly documented the hazards associated with the facility's operations.

A summary of the safety class SSCs and OSRs that provide preventive and/or mitigative features for the various accidents, natural phenomena events, and external events is provided in Table 2. The DOE review effort concluded that the PFP FSAR adequately defined the safety class SSCs necessary for plant safety, and established the required OSRs for safe plant operations.

In short, the DOE review concluded that the risks associated with conducting plutonium handling, processing, and storage operations within PFP facilities, as described in the PFP FSAR, are acceptable, since the accident safety analyses associated with these activities meet the WHC risk acceptance guidelines and DOE safety goals in SEN-35-91.

Table 1. Summary of Accidents, Natural Phenomena Events, and External Events

	Consequences ^(a)	Frequency of Event	Risk Acceptance Guidelines
Accidents			
RMC Line Fire (Section 9.2.2A)	3.6 rem onsite 1.3 rem offsite	Incredible	100 rem onsite 25 rem offsite
Fire in Glovebox MT-5 (Section 9.2.2C)	0.8 mrem onsite 0.3 mrem offsite	Conservatively assumed to be 10^{-2} to 10^{-1} /yr	1 rem onsite 65 mrem offsite
Waste Drum Fire (Section 9.2.2D)	2.2 rem onsite 40 mrem offsite	Conservatively assumed to be 10^{-4} to 10^{-2} /yr	5 rem onsite 500 mrem offsite
Criticality (Section 9.2.3)	0.73 rem onsite 53 mrem offsite	less than 2.5×10^{-2} /yr	1.7 rem onsite 180 mrem offsite
Roof Fire, 234-5Z (Section 9.2.2B)	<i>Radiological consequences are bounded by the RMC Line Fire analysis</i>		
Deflagration in Glovebox MT-5 (Section 9.2.1C)	<i>Radiological Consequences are bounded by the MT-5 fire analysis</i>	Incredible	---
Toxic Chemical Release, CCL ₄ (Section 9.2.8)	150 ppm onsite 1.7 ppm offsite	10^{-4} /yr	232 ppm onsite 29 ppm offsite
Flammable Gas Explosion (Section 9.2.1F)	Not calculated	Incredible	---
Natural Phenomena Events			
Seismic Event, "No-Vent" Case (Section 9.2.4) ^(b)	15.20 rem onsite 0.31 rem offsite	10^{-4} /yr	23 rem onsite 2.8 rem offsite

Table 1 (Cont). Summary of Accidents, Natural Phenomena Events, and External Events

	Consequences ^(a)	Frequency of Event	Risk Acceptance Guidelines
"No-Vent" Case plus Seismic-Induced Fire and Criticality ^(b) (Section 9.2.4.6)	21.25 rem onsite 0.47 rem offsite	10 ⁻⁶ /yr	100 rem onsite 25 rem offsite
Seismic Event, "Vent" Case (Section 9.2.4) ^(b)	12.10 rem onsite 1.91 rem offsite	10 ⁻⁴ /yr	23 rem onsite 2.8 rem offsite
"Vent" Case plus Seismic-Induced Fire and Criticality ^(b) (Section 9.2.4.6)	2.07 rem offsite	10 ⁻⁶ /yr	100 rem onsite 25 rem offsite
232-Z Building Release due to 0.09 g earthquake ^(b)	7.10 rem onsite 0.14 rem offsite	2x10 ⁻³ /yr	8.5 rem onsite 1.0 rem offsite
Seismic-Induced Toxic Release, HNO ₃ (Section 9.2.4A.6.3.5.D)	1.8 ppm onsite 0.032 ppm offsite	10 ⁻⁴ /yr	24 ppm onsite 7.7 ppm offsite
Seismic-Induced Toxic Release, CCL ₄ (Section 9.2.4A.6.3.5.E)	53 ppm onsite 0.98 ppm offsite	10 ⁻⁴ /yr	232 ppm onsite 29 ppm offsite
Seismic-Induced MT-5 Deflagration (Section 9.2.4B)	0.19 mrem onsite 0.07 mrem offsite	Incredible	100 rem onsite 25 rem offsite
Strong Wind, (Section 9.2.5) HNO ₃ release	18 ppm onsite 0.29 ppm offsite	10 ⁻² /yr	19 ppm onsite 3.9 ppm offsite
CCL ₄ release	64 ppm onsite 1.0 ppm offsite	10 ⁻² /yr	100 ppm onsite 17 ppm offsite
External Events			
Aircraft Accident (Section 9.2.7)	Not calculated	Incredible	---

(a) rem is calculated as Effective Dose Equivalent (EDE)

(b) Because earthquakes commonly effect all PFP buildings the release from the 232-Z Building has been included into the totals for both the "No-Vent" and "Vent" cases and is a significant contributor (approximately 40%) to the total release postulated for the Seismic Event. The 232-Z Building is currently scheduled for decontamination and decommissioning.

Table 2. Summary of the Safety Class SSCs and Related OSRs

Safety Class SSCs		Related OSRs
Accidents		
RMC Line Fire (Section 9.2.2A)	Building Structural Features HEPA Filters-Final and Secondary Stages	LCO 3.2.1, <i>Filtered Exhaust From Zones 3 and 4</i> LCO 3.3.1, <i>Control of DISPERSIBLE Plutonium</i> AC 5.20, <i>Fire Protection</i> AC 5.22, <i>Plutonium Inventory</i>
Fire in Glovebox MT-5 (Section 9.2.2C)	Building Structural Features HEPA Filters-Final and Secondary Stages	LCO 3.2.1, <i>Filtered Exhaust From Zones 3 and 4</i> LCO 3.3.1, <i>Control of DISPERSIBLE Plutonium</i> AC 5.20, <i>Fire Protection</i> AC 5.22, <i>Plutonium Inventory</i>
Waste Drum Fire (Section 9.2.2D)	None identified	AC 5.24, <i>Management of Transuranic Material Outside Facility Buildings</i>
Criticality (Section 9.2.3)	Building Structural Features 26 Inch Vacuum Liquid Detection and Interlock Glovebox Sump Disks and Floor Filler Plates HC-6 Raschig Rings & Glovebox Criticality Drains Plutonium Storage Arrays & Criticality Alarm System	LCO 3.1.1, <i>Criticality Prevention System</i> LCO 3.1.2, <i>Criticality Detectors and Alarms</i> AC 5.15, <i>Nuclear Criticality Safety</i>
Roof Fire, 234-5Z (Section 9.2.2B)	None identified	AC 5.20, <i>Fire Protection</i>
Deflagration in Glovebox MT-5 (Section 9.2.1C)	Building Structural Features HEPA Filters-Final and Secondary Stages	LCO 3.2.1, <i>Filtered Exhaust From Zones 3 and 4</i> LCO 3.3.1, <i>Control of DISPERSIBLE Plutonium</i> AC 5.20, <i>Fire Protection</i> AC 5.22, <i>Plutonium Inventory</i>

Table 2 (Cont). Summary of the Safety Class SSCs and Related OSRs

Safety Class SSCs		Related OSRs
Flammable Gas Explosion (Section 9.2.1F)	None identified	AC 5.21, <i>Flammable Gas Inventory</i>
Natural Phenomena Events		
Seismic Event (Section 9.2.4)	Building Structural Features Ventilation System Seismic Shutdown System Plutonium Storage Arrays	LCO 3.1.1, <i>Criticality Prevention System</i> LCO 3.2.3, <i>Ventilation System Seismic Shutdown</i> LCO 3.3.1, <i>Control of DISPERSIBLE Plutonium</i> AC 5.20, <i>Fire Protection</i> AC 5.21, <i>Flammable Gas Inventory</i> AC 5.22, <i>Plutonium Inventory</i>
Seismic-Induced Toxic Release, HNO ₃ & CCL ₄ (Section 9.2.4A.6c-d)	None identified	None identified
Seismic-Induced MT-5 Deflagration (Section 9.2.4B)	Building Structural Features	LCO 3.3.1, <i>Control of DISPERSIBLE Plutonium</i> AC 5.20, <i>Fire Protection</i> AC 5.22, <i>Plutonium Inventory</i>
Strong Wind, HNO ₃ & CCL ₄ (Section 9.2.5)	None identified	None identified
External Events		
Aircraft Accident (Section 9.2.7)	None identified	None identified

2.0 SUMMARY SAFETY ASSESSMENT

A summary of the safety assessment for PFP is provided in Chapter 2 of the PFP FSAR. This chapter was not specifically reviewed since the details of the information contained in this chapter are more thoroughly presented in the other chapters of the FSAR, specifically Chapter 9.0, *Accident Safety Analysis*. Westinghouse Hanford Company did not revise this chapter after significant revisions were made to Chapter 9.0 of the PFP FSAR. A revision should be made to this chapter during the first annual update to make this chapter consistent with chapter 9.0 results. The summary descriptions and conclusions of the various accidents presented in Chapter 2.0 are not necessarily in agreement with those in Chapter 9.0. As a result, only the accident analysis results presented in Chapter 9.0 of the PFP FSAR shall be considered a part of the facility's safety basis and the results presented in Chapter 2.0 shall be considered obsolete.

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3.0 SITE CHARACTERISTICS

A description of the PFP site, its location, nearby population, and land usage is provided in Chapter 3.0 of the PFP FSAR. Westinghouse Hanford Company developed the material in Chapter 3.0 to be consistent with the requirements in Nuclear Regulatory Guide 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*. Although no formal evaluation of the material presented in Chapter 3.0 was performed, the material presented is consistent with the requirements in NUREG 3.39 and is therefore considered adequate.

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4.0 PRINCIPLE DESIGN CRITERIA

Presented in Chapter 4 of the PFP FSAR are the criteria used in design and construction of the PFP facilities. The PFP was designed prior to 1946 and constructed between 1949 and 1982. In each case, the facility was designed in accordance with codes and standards current at that time. The material presented in this chapter of the FSAR resulted from a conformance evaluation of the PFP to 6430.1A, *General Design Criteria*. The conformance evaluation of PFP to DOE Order 6430.1A is provided in WHC-SD-CP-ES-128, *Preliminary Evaluation of PFP Conformance to DOE Order 6430.1A General Design Criteria*.

The review of this chapter included design criteria and plant design for the following subjects:

- Structural and Mechanical
- Structural Response
- Safety Protection Systems
- Safety Class Systems.

These sections were reviewed for compliance to DOE Order 6430.1A and other DOE orders, regulations, and/or documents cited in the sections themselves.

4.1 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

The review of structural and mechanical safety criteria included all of Section 4.2 of the FSAR which discussed seismic loading, wind loading, and other types of loading. One difficulty in addressing seismic and other naturally occurring events was that the original PFP building designs were based upon design codes that did not reflect the more stringent requirements in place today. For example, the Uniform Building Code (UBC) did not include seismic considerations until the 1949 edition. Several of the original PFP buildings were constructed prior to 1949 and in many instances WHC had to employ rigorous analytical techniques to demonstrate that adequate structural capacity existed for seismic loads.

The general findings of the review were that the criteria used by WHC are consistent with the design and evaluation criteria found in DOE Order 6430.1A and were satisfactory. This included all of the naturally occurring events that were addressed in the FSAR with the exception of ashfall. Ashfall loading requirements were also included by WHC and are found to be acceptable. Various acceptable methods of seismic analysis were employed depending on initial investigations coupled with containment requirements of the individual facilities.

4.1.1 Wind Loading (Section 4.2.1)

Discussion. Wind loading criteria are governed by DOE Order 6430.1A. Reference is made by DOE Order 6430.1A to UCRL-15910 for wind loadings on DOE facilities. This includes the design of new facilities as well as the evaluation of existing facilities. The values for the

site wind velocities are given in UCRL-53526 (Coats and Murray 1985). Hanford Plant Standard SDC 4.1 Rev. 11 (Conrad 1989) is used for the design and analysis criteria for facilities at the Hanford Site. It is stated in SDC 4.1 Rev. 11 that the source for wind loading criteria is in UCRL-15910. As a source for guidelines and procedures involving wind loading, ANSI A58.1 is referenced in UCRL-15910 and SDC 4.1 Rev 11.

Evaluation Findings. In Section 4.2.1, "Wind Loading," the PFP FSAR referred to all of the above mentioned documents in discussions of wind loading criteria. The FSAR discussed details about the source of all of the wind loading criteria used in FSAR chapters 5 and 9 and tied those criteria together through DOE 6430.1A. The wind loading criteria stated in the FSAR are consistent with DOE Order 6430.1A and are, therefore, acceptable.

4.1.2 Tornado Loading (Section 4.2.2)

Discussion. Tornado wind loading requirements are addressed in DOE Order 6430.1A that references UCRL-15910 for guidelines to evaluate tornado wind loading; SDC 4.1 Rev 11 also references UCRL-15910 for tornado wind loading. It is stated indirectly by UCRL-15910 and directly by SDC 4.1 that facilities at the Hanford site are to be designed for extreme wind loading and not for tornado loading. Therefore, the evaluation of DOE facilities at the Hanford site does not have to address tornado generated winds.

Evaluation Findings. At the beginning of Section 4.2.2, "Tornado Loadings," the FSAR states that the effects of the design base tornado were not required to be analyzed since extreme wind was the governing wind loading condition at the Hanford site. This was reinforced by guidelines in both SDC 4.1 Rev. 11 and UCRL-15910. These criteria have been reviewed, and WHC's conclusion that tornado wind loading analyses are not required for PFP evaluation is correct.

4.1.3 Water Level (Flood) Design (Section 4.2.3)

Discussion. Based on site elevation with respect to the Columbia River, SDC 4.1 Rev 11 contains the maximum probable flood level requirement at the Hanford site. The highest flood elevation for the maximum probable flood within river miles 340 through 392 along the Columbia River is slightly less than 423 ft above mean sea level. Since the lowest portion of PFP is at approximately 660 ft above mean sea level, no requirements exist to calculate flood loading for the facility. Floods due to heavy rainfall are not mentioned in SDC 4.1 Rev. 11. The calculation in Section 6 of floods due to local rainfall is discussed in UCRL-15910.

Evaluation Findings. In the FSAR, WHC indicated that structural failures due to water flooding are not credible for two separate flooding events. The first event, Columbia River flooding, involved the elevation of PFP with respect to the elevation of the Columbia River. Although the FSAR indicated a flood level of 423 ft above mean sea level at 100-N reactor site, further upstream a level of 450 ft was the maximum level to be concerned about. This slight discrepancy is unimportant since PFP is significantly higher than either of the two flood

elevations stated. Therefore, flooding due to the Columbia River was not considered to be a credible event. The second event considered was flooding due to a heavy rainfall. In Section 3.4.2.3, the FSAR considered the probable maximum precipitation that could be expected to occur in 24 hours at least once in a million years. The probable maximum precipitation was estimated to be 11 inches. Westinghouse Hanford Company dismissed the possibility of flooding due to a heavy local rainfall because the soil around the site is considerably porous and would drain off the probable maximum precipitation that could be expected to occur in 24 hours. Based upon this discussion, the conclusion reached by WHC is correct; there is no need to calculate flood loads during Columbia River flooding or local rainfall flooding.

4.1.4 Missile Protection (Section 4.2.4)

Discussion. Missile protection for a facility is closely related to wind loading and/or tornado loading. As discussed previously, UCRL-15910 discusses missiles generated by extreme winds and by tornado winds. It should be noted that UCRL-15910 indicates that, for the Hanford site, the governing criteria for wind generated missiles is for extreme winds and not for tornado winds. This is based upon the wind-tornado transition speed for which the probability of exceedance is below the threshold value of 1.0×10^{-5} . Therefore, the criteria for evaluation should be based upon UCRL-15910 table 5-2 for wind.

Evaluation Findings. The FSAR for PFP contains the proper criteria in Section 4.2.4, which was also found in SDC 4.1, Rev. 11. However, the FSAR stated that the criteria was for tornado generated missiles. This wording is incorrect and should be restated as extreme wind generated missiles rather than tornado generated missiles. Recommendations for FSAR updates are given in Appendix B. Other than the wording problem, WHC's criteria for missile protection is correct and meets DOE Order 6430.1A.

4.1.5 Seismic Design (Section 4.2.5)

Discussion. This section describes the evaluation of the seismic analysis methods and criteria. To evaluate the seismic analysis performed for the PFP, it is important to understand what analyses and methodologies are presently required. It must be realized that the FSAR preparation for PFP has been a continuous effort throughout the 1980s. Changes in criteria and acceptable methodologies have been instituted by DOE, and additional requirements have been imposed on DOE facilities since the FSAR preparation was started and completed. Also, it is important to note that some of the "earlier" analyses may or may not be more conservative than what is presently required.

Requirements for seismic evaluation of existing DOE facilities are found in DOE Order 6430.1A. This order has effectively categorized facilities based upon a hazard class rating. The four ratings are 1) High Hazard 2) Moderate Hazard 3) Important or Low Hazard, and 4) General Use.

The following two key documents are used extensively for seismic evaluation of existing facilities and design of new facilities, and are also referenced by DOE Order 6430.1A: UCRL-15910 and UCRL-53582 (Murray and Coats 1984). The system of hazard ratings for buildings and requirements for the type of analysis performed for the "graded approach" are presented in UCRL-15910. Figures for the peak accelerations and the design response spectrum for all of the major DOE sites around the country, including the Hanford site, are presented in UCRL-53582.

The requirements for the seismic evaluation are based on a graded approach in which a facility is assigned a hazard rating depending on the facility's use and the amount of radioactive material at the facility. The four categories mentioned above are defined in UCRL-15910. Most of the buildings were evaluated using a dynamic analysis (elastic dynamic or non-linear time history analyses) as required for UCRL-15910 for high hazard facilities. However, a provision exists in UCRL-15910, Appendix A, that allows the use of equivalent static methods for very simple structures.

The peak horizontal ground accelerations or zero period accelerations or "g-load" levels (all three terms are equivalent) used in most of the seismic analyses performed by WHC were higher than what is currently required (0.25-g instead of 0.20-g). The following quote from the FSAR, Section 5.2.1 explains the discrepancy:

"Most of the seismic qualification analyses for the PFP discussed in this chapter were performed in accordance with SDC 4.1, Rev. 10, used a zero period acceleration of 0.25 g. Since the completion of these analyses, Revision 11 has been implemented. In Revision 11, the seismic zero period acceleration is 0.20 g for nonreactor high hazard facilities and Safety Class 1 items. Revision 10 response spectra are based on 84% amplification factors, whereas Revision 11 response spectra are based on median amplification factors."

Evaluation Findings. The independent evaluations of the methods used for individual PFP structures are included in SER Section 9.2.4 since many building analyses had some variations in the methodologies used to determine seismic structural integrity. In general, the PFP seismic evaluations were performed to DOE Order 6430.1A. Other than some minor details that are discussed later in this document, the methodologies and analyses were acceptable. The depth of analysis depended on initial investigations that determined whether each building required more detailed non-linear time integration analyses, or whether the modal expansion techniques were adequate. The depth of the analyses are found to be acceptable.

4.1.6 Snow and Ashfall Loading (Section 4.2.6)

Discussion. Snow loads for DOE facilities are discussed in DOE Order 6430.1A. The DOE order refers to ANSI A58.1 as the source for roof loads, which includes snow loads. Ashfall loading is not mentioned in DOE Order 6430.1A. Although DOE Order 6430.1A references ANSI A58.1 (which is now also called ASCE 7-88) for general roof loading, it also contains

no specific requirements for ashfall loading. However, specific requirements are included in SDC 4.1 Rev. 11. The source for SDC 4.1 ashfall loading was obtained from the Washington Public Power Supply System FSAR (WPPSS 1981). The PFP FSAR ashfall loading criteria that was taken directly from SDC 4.1 Rev. 11 states the following:

"Revision 11 also modifies the ashfall loading requirements for the design of Safety Class 1 structures to be consistent with the latest data generated by WPPSS (Washington Public Power Supply System)."

Evaluation Findings. Concerning requirements for snow loads, FSAR section 4.2.6 quoted ANSI A58.1 snow loading requirements and is acceptable. The loading criteria for ashfall, although not an explicit requirement in DOE Order 6430.1A, is acceptable for the Hanford Site. The WPPSS ashfall loading requirements were found to be acceptable by the U.S. Nuclear Regulatory Commission.

4.1.7 Process and Equipment Derived Loads (Section 4.2.7)

Discussion. DOE Order 6430.1A requires that design dead loads be included in the weight of all permanent service equipment. Service equipment includes plumbing stacks, piping, heating and air-conditioning equipment, electrical equipment, flues, fire sprinkler piping and valves, and similar fixed furnishings. The weight of service equipment that may be removed with change of occupancy of a given area are considered as live loads. Section 4.2.7 of the PFP FSAR stated that actual equipment loads were used in calculating building performance.

Evaluation Findings. The PFP FSAR is consistent with DOE Order 6430.1A and is acceptable.

4.1.8 Combined Load Criteria (Section 4.2.8)

Discussion. Combined load criteria are given in DOE Order 6430.1A. In particular, Section 011-99.0.8 considers load combinations. For safety class structures, UCRL-15910 is to be used as a basis for natural phenomena hazards. For concrete structures, ACI-349 load combinations are used. For steel structures, AISC N690 load combinations are used.

Evaluation Findings. The PFP FSAR refers to the same documents listed above. Furthermore, the FSAR quotes portions of SDC 4.1 for various loadings on structures. With the exception of ashfall loads, the SDC 4.1 load requirements are derived from DOE 6430.1A, ACI-349, AISC N690, ACI-531, and UBC. The PFP FSAR presents specific load combinations in Table 4.2.10.1, which also references the appropriate section in DOE Order 6430.1A from which the load combinations are derived. The load combination criteria listed by WHC are adequate.

4.1.9 Subsurface Hydrostatic Loading (Section 4.2.9)

Discussion. It is stated in ANSI A58.1 that subsurface hydrostatic loading is generally a concern when structures are below grade and when a portion of the surrounding soil is below the water table. The PFP is approximately 200 feet above the water table. Furthermore, the porous soil conditions allow for the rainfall to drain away quickly. The FSAR concludes that credible subsurface hydrostatic loadings do not exist and do not need to be evaluated.

Evaluation Findings. Based upon the above discussion, WHC's conclusion is acceptable.

4.1.10 General Criteria (Section 4.2.10)

Discussion. The general criteria that WHC provides in section 4.2.9 and in Table 4.2.10-1 of the FSAR summarize all of the various design and loading criteria discussed above. All of the criteria listed were then cross referenced to the applicable section in DOE Order 6430.1A and to the PFP FSAR reference sections.

Evaluation Findings. The table provided by WHC that is associated with section 4.2.9 is a helpful guide in determining the source of the various requirements within DOE Order 6430.1A. The general criteria table and statement in the PFP FSAR are found to be acceptable.

4.2 STRUCTURAL SPECIFICATIONS

This section discusses the basis and engineering design required to maintain the confinement integrity of major buildings within the PFP complex. A summary of the structure type, materials of construction, and construction classifications are presented in Table 4.1. Significant natural phenomena hazards analysis, design information, and conclusions specific to the major PFP process buildings are presented in Table 4.2. The natural phenomena hazards and applicable design criteria were presented in the preceding section.

4.2.1 Building 234-5Z

Discussion. The actual design of the 234-5Z Building was developed from the criteria that were in effect from 1943 to 1949. During this period, the Hanford site above ground structures, including the 234-5Z Building, were designed for static vertical live and dead loads. Seismic criteria was not added to the UBC until 1949 and were not included in the 234-5Z Building design criteria because Hanford was considered a low intensity, minimum occurrence earthquake zone. Resistance to lateral wind forces was required based on area wind data available at that time.

The building is approximately 500 ft long and 180 ft wide. The frame is structural steel with an outer sheathing of aluminum panels over rock wool insulation and 16-gauge sheet metal. There are also 8 in thick internal reinforced concrete walls, principally running in the east-

west direction, and two box-type reinforced concrete stairwells. The stairwells extend to the roof; the reinforced concrete walls stop at the second floor.

A dynamic seismic analysis was performed on the 234-5Z Building structure. The structure was also evaluated for its response to wind generated wall pressures and missiles per SDC 4.1, Rev. 11. Analyses of the 234-5Z Building show that the main load bearing structure is not over-stressed by the extreme wind or adversely impacted by missile loads. The external skin panels were tested and found acceptable for missile resistance. The effects of both live and dead loads, and snowfall, were found to be acceptable per ANSI A58.1 and SDC 4.1, Rev. 11 guidelines.

Evaluation Findings. The seismic, wind, and snow loadings on the 234-5Z Building have been reviewed. A review of the seismic analysis for this building is presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5. The criteria of live, dead, snowfall, and seismic loads for this building were determined to be in accordance with SDC 4.1, Rev. 11 and ANSI A58.1 and are acceptable.

4.2.2 236-Z Building

Discussion. This building is essentially a four story structure, 79 ft x 71 ft x 47.5 ft, surmounted at the southwest corner by a two story penthouse. The 236-Z Building was designed to 1961 UBC standards, which included a seismic criterion (0.10 g) that is significantly lower than the present day criterion (0.20 g). With the exception of the roof, the south end of the process cell, and the fourth floor ceiling, the building is constructed of reinforced concrete. The roof is open web steel joist framing, steel decking with rigid insulation consisting of light weight concrete fill and gravel covered roofing. A portion of the south wall is also the 1 ft thick wall of the process cell and includes an opening in the reinforced concrete wall (this opening is filled by a large steel door surrounded by a concrete block wall). The south wall has a bridge shaped cross section of concrete blocks, and recently has had welded steel plates added to preclude leakage during a seismic event.

Seismic analyses and evaluations were performed for the 236-Z Building to determine whether it could resist the DBE. Three-dimensional lumped mass beam models of the building were developed. Seismic forces and moments for these models were computed using the modal superposition response spectra method of dynamic seismic analysis. The seismic evaluation indicates that the 236-Z Building meets requirements in SDC 4.1, Rev. 11. Analyses of the 236-Z Building show that the main load bearing structure is not overstressed by extreme wind or adversely impacted by missile loads. Also, the effects of live and dead loads, and snowfall were found acceptable per ANSI A58.1 and SDC 4.1, Rev. 11 guidelines.

Table 4.1. PFP Building Construction Summary

Building	Structure	Roof	Exterior Walls	Floors
234-5Z	Braced steel frame	Metal deck	Corrugated metal	Concrete
236-Z	Reinforced Concrete	Concrete slab over metal deck	Concrete	Concrete
232-Z	Reinforced concrete block	Concrete slab over metal deck	Concrete block	Concrete
241-Z (below grade)	Reinforced concrete	Reinforced concrete	Reinforced Concrete	Reinforced concrete
241-Z (above grade)	Corrugated steel	Corrugated steel	Corrugated steel	Reinforced concrete
242-Z	Braced steel frame	Metal deck	Corrugated metal	Concrete
291-Z	Reinforced concrete	Concrete	Concrete	Concrete
291-Z-1 Stack	Reinforced concrete	N/A	N/A	N/A
2734-ZL	Corrugated steel	Corrugated steel	Corrugated steel	Concrete
2721-Z	Reinforced concrete	Concrete	Concrete	Concrete
2736-Z	Reinforced concrete	Concrete	Concrete	Concrete
2736-ZA	steel and concrete	Concrete	Concrete	Concrete
2736-ZB	steel and concrete	Concrete	Concrete	Concrete

Table 4.2. Natural Phenomena Hazards

Building	Seismic Analysis Method	Wind Loads	Snow ⁽¹⁾ Loads	Conclusions
234-5Z	Dynamic	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow
236-Z	Dynamic	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow
232-Z	static	none	load included in seismic analysis	Building withstands a 0.09 g earthquake, failure assumed at larger loads (see SER Section 9.2.4)
241-Z	none	SDC 4.1	SDC 4.1 ANSI A58.1	Cover blocks withstand DBE, structure failure assumed in SER Section 9.2.4
242-Z	Dynamic	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE with modifications (see SER Section 9.2.4), can withstand wind, missile, and snowfall
291-Z	Dynamic	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow
291-Z-1 stack	Dynamic	SDC 4.1	Not applicable	stack can withstand DBE, wind, missile, and snow
2721-Z	static	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow
2734-ZL	static	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, and snow. Missile penetration assumed in Chapter 9
2736-Z	Equivalent Static	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow
2736-ZA	Equivalent Static	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow
2736-ZB	Equivalent Static	SDC 4.1	SDC 4.1 ANSI A58.1	can withstand DBE, wind, missile, and snow

(1) Analyses for combined loading of snow plus ashfall and for ashfall only were not performed since these are required only for new safety class 1 facilities per SDC 4.1, Rev. 11.

Evaluation Findings. The seismic, wind, snow loadings on the 236-Z Building have been reviewed. A review of the seismic analysis for this building is presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5. The effects of live and dead loads, and snowfall analyses performed for this building were determined to be in accordance with SDC 4.1, Rev. 11 and ANSI A58.1 and are acceptable.

4.2.3 232-Z Building

This building houses what was previously the incinerator facility. This facility is no longer in service and is currently scheduled for decontamination and decommissioning. It is constructed out of reinforced concrete blocks. It is approximately 37 ft wide by 57 ft long. The building has a single story over the process areas and two stories over the service areas at the north end of the building. The respective roofs are 15 ft and 19 ft above grade and consist of concrete over a metal decking, with insulation and a built-up asphalt gravel covering. Westinghouse Hanford Company performed equivalent static analysis to demonstrate that the facility would not collapse at seismic loads up to 0.09 g. At greater than 0.09 g, the facility is assumed to collapse. A portion of the residual plutonium inventory remaining in the building is assumed to be released to the environment during a seismic event greater than 0.09 g. The acceptability of the building failure and consequences are discussed in SER Section 9.2.4.4g.

4.2.4 241-Z Building

The 241-Z Building is designated as the Waste Retention Facility. It is a buried structure with its roof at grade level. It consists of five separate enclosures or cells which contain a 4,200 gallon tank. The tanks are used to accumulate PFP liquid waste before transfer to underground waste storage tanks. It is built of reinforced concrete. The building cover is pre-engineered corrugated steel.

Seismic analysis of the below grade reinforced concrete vault has not been documented. However, seismic analyses have been performed on the reinforced concrete vault cover blocks which shows that they will withstand a DBE. Although the tanks can fail during the DBE, cover blocks will not fall into the cells and cause further damage. The acceptability of the building and tanks failure is discussed in SER Section 9.2.4. Analyses of extreme wind and associated missile hazards and snow loading show that the design capacity of the building is less than design criteria in SDC 4.1, Rev. 11 and ANSI A58.1. Therefore, the above ground structures are assumed to fail in the event of these natural phenomena. The consequences of these events are bounded by the seismic analysis discussed in SER Section 9.2.4.

4.2.5 242-Z Building

Discussion. The dimensions of the building are 40 ft x 26 ft x 23 ft. The south wall is constructed of reinforced concrete. The remainder of the building has a structural steel frame covered with metal lath and plaster internally, and insulating material wall panels externally.

The roof is constructed of metal decking, covered by insulation and built up asphalt and gravel.

A dynamic seismic analysis was performed to determine building frequencies. The analysis showed that the stresses of some members are not within code allowables. During a DBE, structural failure is assumed for these members. The acceptability of the failure and the consequences are discussed in SER Section 9.2.4. The results of the extreme wind and associated missiles analyses show that the main load bearing structures can withstand the extreme wind forces and the wind generated missiles do not penetrate the wall panels. Also, the effects of live and dead loads, and snowfall were found to be within code allowables per ANSI A58.1 and SDC 4.1, Rev. 11 guidelines.

Evaluation Findings. The seismic, wind, snow loadings on the 242-Z Building have been reviewed. The results of the seismic analysis for this building are presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5. The effects of live and dead loads, and snowfall analyses performed for this building were determined to be in accordance with SDC 4.1, Rev. 11 and ANSI A58.1 and are acceptable.

4.2.6 291-Z Building

Discussion. The 291-Z Building, known as the exhaust fan house, exhaust air stack building, or compressor and fan house, is a reinforced concrete structure. The building is of irregular shape with dimensions of 74 ft x 143 ft x 23 ft, with only 4 ft above grade.

A detailed seismic analysis was performed for this building which verified that the building will maintain confinement of radioactive materials in the event of a DBE. It was also determined that the integrity of the walls will be maintained in the event of extreme winds and associated missiles. Also, the effects of live and dead loads, and snowfall were found to be within code allowables per ANSI A58.1 and SDC 4.1, Rev. 11 guidelines.

Evaluation Findings. The seismic, wind and associated missiles, and snow loadings on the 291-Z Building have been reviewed. The results of the seismic analysis for this building are presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5. The effects of live and dead loads, and snowfall analyses performed for this building were determined to be in accordance with SDC 4.1, Rev. 11 and ANSI A58.1 and are acceptable.

4.2.7 291-Z-1 Stack

Discussion. The 291-Z-1 stack is located adjacent to the 291-Z Building. The stack and the building are separated by an expansion joint. Because the stack is structurally separated from the 291-Z Building, it was evaluated independently. The stack is constructed of reinforced concrete and is 200 ft tall. The stack has an interior diameter varying from approximately 13½ ft at the top to 16 ft at the base. The thickness of the wall varies with height from 6 in.

at the top, to 9 in. at the base. The foundation of the stack is a massive concrete footing octagonal in shape, with a dimension of 32 ft across its flat sides and 27 ft thick. A 16 ft diameter elbow opening in the footing connects the bottom of the stack to the duct of the 291-Z Building.

A seismic analysis was performed to evaluate the capacity of the stack to resist the DBE. The dynamic analysis demonstrated that the stack has enough ultimate ductile capacity to resist the DBE loading without collapse. The wind analysis showed that the moments and shears generated by the extreme wind and associated missiles is significantly less than the stack capacity and the seismic demand at all elevations.

Evaluation Findings. The seismic and extreme wind and associated missiles generated loads on the 291-Z-1 stack have been reviewed. The results of the seismic analysis for this building are presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5.

4.2.8 2736 Building Complex

Discussion. The 2736 building complex is composed of the 2736-Z, 2736-ZA, and 2736-ZB Buildings. They are one story buildings constructed of reinforced concrete walls and roofs supported by cast-in-place concrete columns.

The 2736-Z Building structure was modeled as a single degree-of-freedom system with the roof dead load plus 25% live load plus half the weights of the walls all concentrated at the roof level. Lateral stiffness was assumed to be provided by building walls, combinations of tilt-up precast panels and cast-in-place reinforced concrete columns, acting as shear walls over their full length. The evaluation confirmed that the building could withstand the current design criteria SDC 4.1, Rev. 11. The 2736-ZA and 2736-ZB buildings were also shown to meet the requirements of SDC 4.1, Rev. 11 seismic criteria. It was also shown that the buildings could withstand the effects of extreme winds and associated missiles and roof loadings per the criteria in SDC 4.1, Rev. 11 and ANSI A58.1.

Evaluation Findings. The seismic, wind and associated missiles, and snow loadings on the 2736 building complex have been reviewed. The results of the seismic analysis for this complex are presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5. The effects of live and dead loads, and snowfall analyses performed for these buildings were determined to be in accordance with SDC 4.1, Rev. 11 and ANSI A58.1 and are acceptable.

4.2.9 2721-Z Building

The 2721-Z Building is a one story structure constructed of poured reinforced concrete, with 1 ft thick walls. The building was designed and constructed to withstand seismic and tornado wind loads described in SDC 4.1, Rev. 7 (0.25 g ZPA, 175 mi/h). Therefore, the building

exceeds the current requirements of SDC 4.1, Rev. 11, (0.2 g ZPA, 90 mi/h) for seismic, wind, and wind missile resistance. Also, the effects of live and dead loads, and snowfall loads on the roof were found to be within code allowables per ANSI A58.1 and SDC 4.1, Rev. 11 guidelines.

4.2.10 2734-ZL Building

Discussion. Building 2734-ZL is a prefabricated structure constructed out of corrugated sheet metal sidings and roof with dimensions of 12 ft x 12 ft x 8 ft. The building was analyzed to SDC 4.1, Rev. 11 requirements. The seismic analysis demonstrated that the building can withstand the DBE. The analysis for extreme wind verified that the building can withstand wind loads. However, it is assumed that the building walls can be penetrated by a wind driven missile. The consequences of this event are discussed in SER Section 9.2.5. The effects of live and dead loads, and snowfall loads on the roof were found to be within code allowables per ANSI A58.1 and SDC 4.1, Rev. 11 guidelines.

A large duct which extends from the 234-5Z Building to the 291-Z Building travels directly above the 2734-ZL Building. This duct was analyzed for seismic and snow loadings. The results of these evaluations demonstrated that the duct can withstand the DBE and snow loads.

Evaluation Findings. The seismic, wind and associated missiles, and snow loadings on the 2734-ZL Building and the large duct directly above it have been reviewed. The results of the seismic analysis for this building and duct work are presented in SER Section 9.2.4. A discussion of wind and missile hazards are presented in SER Section 9.2.5. The effects of live and dead loads, and snowfall analyses performed for this building and duct work were determined to be in accordance with SDC 4.1, Rev. 11 and ANSI A58.1 and are acceptable.

4.3 SAFETY PROTECTION SYSTEMS

The safety protection systems that define a portion of the facility's safety envelope are discussed in this section. These systems, structures, and components (SSCs) include structural features and ventilation components which comprise the confinement system, criticality alarm systems, and nuclear criticality controls.

4.3.1 Protection by Multiple Confinement Barriers and Systems

Physical barriers (e.g., facility layout, primary and secondary process equipment systems and components, enclosures, ventilation and filtration systems) are referred to as confinement systems. Physical confinement barriers must be established to prevent uncontrolled release of radioactive material to the environment and mitigate the spread of contamination within the facility. Barriers between the environment and areas of a nuclear facility that contain and/or have the potential to contain radioactive material are required by DOE Orders. This section of the PFP FSAR provides the applicable criteria selected from DOE Order 6430.1A, *General Design Criteria*, and other referenced documents required to design, construct, and implement

confinement systems in the PFP. This review included all of section 4.3.2 and various sections of Chapter 5 of the PFP FSAR that discussed confinement barriers and Systems and Ventilation/Offgas Systems. The general findings of the review were that the design and evaluation criteria meet DOE Order 6430.1A and are satisfactory.

4.3.1.1 Confinement Barriers and Systems (Section 4.3.2.1)

Discussion. The principle of confinement in the PFP facilities features a system of primary, secondary, and tertiary confinement boundaries to prevent a release of radioactive material to the environment. These confinement boundaries are defined as follows:

- Primary Confinement - Primary confinement shall consist of barriers, enclosures, gloveboxes, piping, vessels, tanks, etc. that contain plutonium. The principle function of the primary confinement is to prevent release of plutonium to areas other than where processing operations are normally conducted.
- Secondary Confinement - Secondary confinement shall consist of barriers and associated ventilation systems that confine potential release of hazardous material from primary confinement. Because plutonium processing in the PFP is normally conducted in gloveboxes as the primary confinement, the operating area boundary and the ventilation system serving that operating area serve as the secondary confinement system.
- Tertiary Confinement - Tertiary confinement shall be provided by the building walls and other outer structures of the facility. For some design basis accidents (DBAs), it represents the final barrier to the potential release of hazardous material to the environment; for others (such as high wind loads), it is the barrier protecting the rest of the facility from damage.

Current design criteria for this facility are defined in DOE Order 6430.1A, RL Order 5480.1A, *Environment, Safety, and Health Program for DOE Operations at RL*, and the sections of WHC-CM-4-9, "Radiological Design," pertaining to confinement barriers and systems. The RL Order is site specific and is derived from the general DOE Order 5480 series. These orders contain specific criteria for

- confinement barriers
- design criteria for enclosures and equipment
- the use of As Low As Reasonably Achievable (ALARA) objectives to ensure that exposure to the public, and to the workers, is minimized.

Evaluation Findings. Applicable design criteria for confinement barriers, process equipment, and enclosures have been reviewed. The scope of the review has included comparisons of PFP design criteria to that required by DOE Order 6430.1A. Applying RL Order 5480.1A and WHC-CM-4-9 in some cases were found to contain more stringent criteria than

comparable criteria from DOE Order 6430.1A. Since the confinement barriers and system design criteria stated in the PFP FSAR were found to be consistent with DOE Order 6430.1A, they are acceptable.

4.3.1.2 Ventilation—Offgas (Sections 4.3.2.2 and 5.4.1)

Discussion Findings. Current design criteria for the facility ventilation system are defined in DOE Order 6430.1A and the sections of WHC-CM-4-9, "Radiological Design," pertaining to ventilation systems. This section of the FSAR contains selected criteria for ensuring capacity standards for normal and abnormal conditions, and ensuring continuity of operation under accident conditions and special control instrumentation. The criteria for the design of ventilation and offgas systems include

- airflow patterns and velocity with respect to contamination control
- minimum negative pressures at key points in the system to maintain proper flow control
- interaction of offgas systems with ventilation systems,
- minimum filter performance with respect to particulate removal efficiency and maximum pressure drop
- minimum performance of other radioactivity removal equipment
- minimum performance of dampers and instrumented control
- ensured continuity of operations under all credible operating conditions.

In general, the PFP facility is divided into zones as a means of control to ensure the confinement of radioactive materials. A separate filtered ventilation exhaust system exists for each zone. Pressure differentials are maintained in the ventilation system between the zones to assure that airflow is from the lowest potential contamination areas to intermediate potential contamination areas to highest potential contamination areas. The intermediate potential contamination area (collectively referred to as Zone 3 in the 234-5Z, 232-Z, 236-Z, 241-Z, and 242-Z Buildings) consists of areas in which radioactivity is stored or handled in contained form, and where a potential for contamination exists. The high potential contamination areas, referred to as Zone 4, consist of the insides of hoods, gloveboxes, and process cells directly exposed to plutonium.

Radioactive materials within process areas are confined as close to the point of origin as practical through using HEPA filters installed on the exhaust ducts of hoods and both supply and exhaust ducts of gloveboxes. Exhaust ventilation air from Zone 3 areas is filtered via one stage of testable HEPA filtration before discharge to the atmosphere through a stack. Air from exhaust Zone 4 is filtered by at least two stages of testable HEPA filtration before discharge to atmosphere via a stack. The HEPA filters (except exhaust filters on gloveboxes) are tested annually. Filters must meet 99.95% efficiency criteria (Surveillance Requirement SR 3.2.1.1.1). Redundant filters or filter banks are provided such that failed filters may be isolated for replacement with continued filtration service provided by installed spares.

The HEPA filters, as they are received onsite, are tested by Hanford Environmental Health Foundation personnel to assure that 99.95% of 0.3 μm diameter particles are retained. The HEPA filters are then retested before installation to ensure particle retention and integrity. Once in service, filters are routinely tested per OSR requirements. Installed pressure instrumentation measures the pressure drop across the filters to assure that the filters are not plugged or breached. Filters are replaced that do not retain 99.95% of the less than 0.3 μm diameter test particles during field testing.

Redundant capacity is provided for key features of the ventilation systems (supply and exhaust fans and HEPA filtration rooms). Two independent 13.8 kV electrical power lines (C8-L1 and C8-L2) supply normal power to the PFP facilities. Four supply fans and four exhaust fans are supplied from circuit C8-L1, and four supply fans and three exhaust fans are supplied from circuit C8-L2. If one circuit fails, manual switching can be done to power all supply and exhaust fans from the remaining circuit. Therefore, failure of either circuit would not result in failure of the ventilation system. In the event of a loss of normal electrical power, two additional ventilation exhaust fans powered by steam turbines are available to provide emergency exhaust.

Detailed accident analyses for the PFP have not identified any scenarios where continued operation of the ventilation system is essential to prevent an unacceptable airborne release of plutonium. Sections 9.1.7 and 9.2.2 of the FSAR provide a comprehensive overview of the ventilation system as it relates to safety issues. Therefore, a static air posture is an acceptable facility situation. This finding resulted in most ventilation system equipment being assigned as either Safety Class 2 or 3. The final stage of HEPA filtration, the stack air monitors, and the seismic cutoff switch are the exceptions, and are considered Safety Class 1.

Evaluation Findings. Applicable design criteria for the ventilation offgas systems have been reviewed. The scope of the review has included comparisons of PFP design criteria to that required by DOE Order 6430.1A. Applying WHC-CM-4-9 in some cases was found to contain more stringent criteria than comparable criteria from DOE Order 6430.1A. Since the ventilation and offgas systems design criteria, as stated in the PFP FSAR, were found to be consistent with DOE Order 6430.1A, they are acceptable.

The results of ventilation upset conditions are discussed in SER Sections 9.1.7 and 9.2.2. These analyses demonstrate that the ventilation system meets the requirements of the design criteria and is acceptable.

4.3.2 Nuclear Criticality Safety (Section 4.3.4)

This section describes the criteria used for establishing acceptable criticality safety controls (CSC) and outlines the engineering and administrative controls used to prevent criticality accidents.

Discussion. Westinghouse Hanford Company references four different sources for design criteria: DOE Order 6430.1A; DOE Order 5480.5; WHC-CM-4-29, Nuclear Criticality Safety Manual; and the American Nuclear Society (ANSI/ANS standards) 8 series of standards on nuclear criticality safety. The FSAR includes a list of general design criteria used for criticality safety at PFP. The FSAR lists several types of controls used in criticality safety that include the following items: mass, volume, geometry, density, area density, enrichment, moderation, material compounds and forms, piece-count, interaction, reflection, absorber, flow, placement or displacement, and chemistry. Also, the FSAR describes the error and contingency criteria used at PFP. The FSAR includes a section on verification of the problem, validation of the computer codes, and review of the criticality safety evaluation.

Evaluation Findings. The references used as a basis for CSCs are acceptable. The design criteria presented are essentially verbatim from DOE Order 6430.1A. The CSCs are standard and are acceptable. The description of the error and contingency criteria have been used commonly in the past and are acceptable. The verification, validation, and review methods reference ANSI/ANS standards 8.1 and 8.19 for the details of the methods and are acceptable.

4.3.2.1 Control Methods for Prevention of Criticality (Section 4.3.4.1)

Two separate discussions and evaluations of CSC are addressed. The first, a methods section, addresses the physical methods of control. The second is a discussion of definitions of a contingency. The separation was made to allow for a more detailed discussion in the contingency definition.

Discussion. The primary CSCs include:

- limiting geometry of containers and process components used to handle fissile material
- limiting mass of material in a specific location
- maintaining spacing of containers containing material
- limiting moderators mixed with fissile material.

Evaluation Findings. The methods used to control criticality are consistent with DOE Order 6430.1A Division 1300 Section 4 and DOE Order 5480.5 Section 12. The CSCs are appropriately applied in the facility.

4.3.2.2 Definition of Two Contingency Criticality Control (Section 4.3.4.2)

Discussion. The PFP processes are designed using the two contingency policy that requires two contingencies to be simultaneously violated for a criticality accident to occur. This policy requires two unlikely and independent errors to occur concurrently before an accidental criticality could occur. Westinghouse Hanford Company uses the concepts of contingency, unlikely event, and worst foreseeable event interchangeably. Westinghouse Hanford Company often considers a contingency to be a combination of two or more errors.

Evaluation Findings. The PFP processes are designed using the two contingency policy that requires two contingencies to be violated in order to allow a criticality; this is an acceptable approach. Using a combination of errors to define a contingency is acceptable. Westinghouse Hanford Company uses qualitative arguments to demonstrate that a contingency is unlikely. Westinghouse Hanford Company's methodology and construction of the qualitative arguments for demonstrating the frequency of a contingency control is not as formal as other methods used in the DOE complex. However, the arguments used to develop the two contingency safety are acceptable.

4.3.2.3 Methods of Criticality Analysis (Section 4.3.4.3)

Discussion. A wide variety of methods are used for criticality analysis. The methods may include Monte Carlo calculations, surface density analogues, comparison to well-referenced data, prior analysis for DOE, and the Department of Transportation (DOT). To model various systems for analysis, WHC often makes simplified assumptions about Pu-240 content (0% Pu-240), moderation (optimally moderated), and reflection (full water or concrete).

Evaluation Findings. Westinghouse Hanford Company's methodology for criticality analysis is appropriate. Specific application of the methods or computer calculations are not in the FSAR and were not reviewed.

The modeling methods and assumptions described in the FSAR are consistent with standard criticality methods used in industry. The methods described include appropriate searches on significant criticality parameters. The assumptions described in the FSAR for Pu-240 content, moderation, and reflection are conservative.

4.4 FIRE AND EXPLOSION PROTECTION (Sections 4.3.6 and 5.4.11)

This section describes the criteria used for establishing the acceptability of the fire protection systems and outlines the engineered and administrative controls used to prevent fires and explosions.

Discussion. The original design criteria for the PFP fire protection systems were designed to minimize the spread of contamination. The application of fire extinguishing liquids or gases by suppression systems were designed to avoid Pu displacement as well. The fire protection philosophy in the PFP facility buildings used to process, handle, or store products, minimizes the amount of combustible material within the structures and in the operation of the facility. Administrative Control (AC) 5.20 requires a fire protection program to be implemented at PFP. Key elements of the program include minimizing combustible loading of facilities, maintaining and periodic testing of the fire protection systems, and control of fire protection system repairs.

Fire and explosion protection design criteria are addressed in DOE 6430.1A. Additional criteria presented in DOE 5480.7 define criteria for an "improved risk" level of fire

protection. These criteria are established to reduce the threats to the public health and welfare from fire and explosions.

Evaluation Findings. Applicable design criteria for fire and explosion protection have been reviewed. The scope of the review has included comparisons of PFP design criteria to that required by DOE Order 6430.1A and DOE Order 5480.7. Section 4.3.6 of the FSAR lists the primary design criteria. This list is sufficiently complete and is found to be consistent with DOE Order 6430.1A.

The PFP FSAR indicates that the facility does not fully comply with requirements established in DOE Order 5480.7 for an improved risk facility. The FSAR indicates that the facility lacks a fully automatic sprinkler protection system on the duct level, and that firewall separations throughout the building do not meet the requirements for rated fire barriers. Compliance with the requirement for an independent source of fire water is also in question.

Installation of a sprinkler system in the duct level, fire doors and barriers, as well as improvements in egress routes, were completed in 1993 by WHC Project B-631, "Life Safety Upgrade." The facility is serviced by two independent water lines. Water is supplied to these lines from the multiple pump and reservoir station (located near the West area power plant) which in turn receives water from the Columbia River via a separate pumping station. Westinghouse Hanford Company Project B-604 is currently underway to provide a large reservoir to enhance the fire suppression water supply in the West area and is expected to be completed in October 1996.

Since the fire and explosion protection design criteria stated in the PFP FSAR were found to be consistent with DOE Orders 6430.1A and 5480.7, and the non-compliance issues have been or are scheduled to be corrected, the design of the PFP fire protection system is acceptable.

4.5 SAFETY CLASS SYSTEMS

Discussion. The classification characteristics discussed in FSAR Section 4.4.2 are slightly different from the criteria that WHC has recently established in Appendix C of WHC-SD-CP-OSR-010, *Plutonium Finishing Plant Operational Safety Requirements*, for selecting PFP Safety Class SSCs. During the most recent revision to this section WHC updated their list of Safety Class SSCs in Table 4.4-2, however, they did not change the text in that section to be consistent with the OSR document. Since the most current criteria is defined in Appendix C of WHC-SD-CP-OSR-010, that criteria will be discussed. Westinghouse Hanford Company should update the information presented in Section 4.4 to be consistent with the OSRs.

In WHC-SD-CP-OSR-010, WHC establishes criteria for selecting Safety Class 1, 2, and 3 SSCs. (Note: WHC refers to Safety Class SSCs as Safety Envelope Design Features.)

Safety Class 1 and 2 SSC selection criteria:

1. Structural confinement boundaries necessary for prevention or mitigation of releases of radioactive materials outside of the facility,
2. Ventilation confinement boundaries necessary for prevention or mitigation of releases of radioactive materials outside of the facility,
3. Equipment features, such as that to prevent explosions, that prevent damage to structural or ventilation confinement or is used to maintain or monitor the integrity of the ventilation confinement boundary,
4. Equipment installed to provide an essential barrier utilized in the double contingency principle for prevention of nuclear criticality, and
5. Alarm systems used to evacuate personnel in the event of a nuclear criticality.

Safety Class 3 SSC selection criteria:

6. Internal confinement provided by established facility ventilation zones to prevent the spread of contamination within the facility; systems related to fire protection; and systems for explosion prevention. Failure of these systems could result in grossly jeopardizing facility worker safety.
7. Equipment used to warn personnel of airborne contamination in facility working areas, and
8. Equipment provided for safe evacuation of facility personnel in case of emergency.

In general, Safety Class 1 and 2 SSCs are relied upon in the safety analysis to prevent or mitigate adverse health effects to the public and co-located worker which may result from the release of hazardous material in the event of an accident. Safety Class 3 SSCs protect the facility workers by ensuring confinement of hazards materials, and in the event of a release, notification of an abnormal condition which may warrant evacuation from the facility. Safety Class 3 SSCs are subject to standard design, fabrication, and industrial test standards and codes.

Evaluation Findings. A list of PFP Safety Class 1 and 2 SSCs are presented in Table 4.3. The list was developed based on the above criteria applied to the accident safety analysis results in FSAR Chapter 9.0. Westinghouse Hanford Company's classification of Safety Class 1, 2, and 3 SSCs is consistent with the criteria for safety class items in DOE Order 6430.1A, and is therefore acceptable.

Table 4.3. Safety Class 1 and 2 SSCs

Structures, Systems, and Components	Location	Applicable OSRs
Safety Class 1		
Building Structural Features	232-Z, 234-5Z, 236-Z, 242-Z, 291-Z-1, 2736-Z, and 2736-ZB	AC 5.10
HEPA Filters -- Final and Secondary Stages	234-5Z, 236-Z, 241-Z, 291-Z, 2736-ZA, and 2736-ZB	LCO 3.2.1.1 LCO 3.2.1.2 SR 3.2.1.1.1 SR 3.2.1.2.1
Stack Continuous AirMonitoring Equipment	291-Z-1 Stack, 291-Z-1 Building 296-Z-3 Stack, 241-Z Building 296-Z-5 Stack, 2736-ZB Building 296-Z-6 Stack, 2736-ZA Building	LCO 3.2.2 SR 3.2.2.1 SR 3.2.2.2 SR 3.2.2.3 SR 3.2.2.4
Ventilation System Seismic Shutdown Switch	234-5Z	LCO 3.2.3 SR 3.2.3.1 SR 3.2.3.2
26 Inch Vacuum Liquid Detection and Interlock System	234-5Z	LCO 3.1.1.1 SR 3.1.1.1.1 SR 3.1.1.1.2
Glovebox Sump Disks and Floor Filters	234-5Z	LCO 3.1.1.3 SR 3.1.1.3.1
HC-6 Raschig Rings	234-5Z	LCO 3.1.1.3 SR 3.1.1.3.1
Glovebox Criticality Drains	234-5Z, 236-Z	LCO 3.1.1.2 SR 3.1.1.2.1
Plutonium Storage Arrays	All Buildings	

Table 4.3 (Continued). Safety Class 1 and 2 SSCs

Structures, Systems, and Components	Location	Applicable OSRs
Safety Class 2		
Criticality Alarm System	234-5Z, 236-Z, 241-Z, 2736-Z, and 2736-ZB	LCO 3.1.2 SR 3.1.2.1 SR 3.1.2.2 SR 3.1.2.3 SR 3.1.2.4 SR 3.1.2.5
HEPA Filters -- Final Stage	232-Z	LCO 3.2.1.1 LCO 3.2.1.2 SR 3.2.1.1.1 SR 3.2.1.2.1
296-Z-14 Stack Monitoring Equipment	232-Z	LCO 3.2.2 SR 3.2.2.1 SR 3.2.2.2 SR 3.2.2.3 SR 3.2.2.4

4.6 REFERENCES

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5.0 PLANT DESIGN

A description of the design and usage of the various facilities which comprise PFP is provided in Chapter 5.0 of the PFP FSAR. Westinghouse Hanford Company developed the material in Chapter 5.0 to be consistent with the requirements in Nuclear Regulatory Guide 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*. Although no formal evaluation of the material presented in Chapter 5.0 was performed, the material presented is consistent with the requirements in NUREG 3.39 and is therefore considered adequate.

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6.0 PROCESS SYSTEMS

A description of the facility's various chemical processes, systems, and control instrumentation and the identification of hazards associated with the various facility processes is provided in Chapter 6.0 of the PFP FSAR. Westinghouse Hanford Company developed the material in Chapter 6.0 to be consistent with the requirements in Nuclear Regulatory Guide 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*. Although no formal evaluation of the material presented in Chapter 6.0 was performed, the material presented is consistent with the requirements in NUREG 3.39 and is therefore considered adequate.

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7.0 WASTE CONFINEMENT AND MANAGEMENT

Presented in Chapter 7.0, *Waste Confinement and Management*, of the PFP FSAR are the facility's systems and controls for managing and confining radioactive wastes generated at PFP. The material presented in Chapter 7.0 was developed in accordance with NUREG 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*. Acceptance criteria for Chapter 7 of the PFP FSAR are derived from DOE Order's 5400.5 *Radiation Protection of the Public and the Environment*, 5480.3 *Safety Requirements for Packaging and Transportation of Hazardous Material's, Hazardous Substances, and Hazardous Waste*, 5820.2A *Radioactive Waste Management*, and guidance obtained from DOE Order 5480.23 *Nuclear Safety Analysis Reports*. This review provides a brief overview of the chapter's content, followed by an evaluation section that addresses specific inadequacies that were found to exist in the chapter.

7.1 DISCUSSION

Specific topics presented in FSAR Chapter 7.0 are as follows:

- Waste Management Criteria - Discusses the various criteria by which airborne effluents, liquid wastes, and solid wastes are controlled and monitored.
- Radiological Wastes - Provides a description of the sources of radioactive gaseous effluents and liquids present in the facility. Also discussed are instrumentation and controls, sampling, analysis, and monitoring.
- Nonradiological Wastes - Describes major nonradiological releases of concern (i.e., carbon tetrachloride, hydrogen fluoride gas etc.) including emission levels and controls.
- Ventilation Exhaust Systems - Describes the facility's ventilation exhaust and offgas treatment systems.
- Liquid Waste Treatment and Retention - Discusses the design objectives for the facility's liquid waste treatment systems and system features related to instrumentation, controls, and safety.
- Liquid Waste Solidification - Simply states that the facility does not reduce or solidify liquid waste generated at the plant.
- Solid Wastes - Describes the packaging, storage, and disposal of solid wastes originating from the PFP facility.

- Items Requiring Further Development - Discusses future projects and projects in progress that will improve the facility's ability to manage wastes generated at the plant.

7.2 EVALUATION FINDINGS

Westinghouse Hanford Company developed the information in this chapter to be consistent with the requirements in Nuclear Regulatory Guide 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*. The review of this chapter consisted of ensuring that the information presented in the chapter was consistent with that required by NUREG 3.39. Except for the specific deficiencies discussed below the material presented in Chapter 7.0 of the FSAR is adequate.

Section 7.1.1.1, "Standards for Control of Airborne Effluents," specifies Environmental Control Limits that were promulgated prior to the time the FSAR was issued (1/91). The EPA has significantly changed and added to the requirements in 40 CFR 61 (NESHAPS) related to the Clean Air Act. The new regulations dated 12/89 now specify that the radiation dose to the public from all pathways associated with releases to the atmosphere should not exceed 10 mrem/yr effective dose equivalent (EDE). The limit applies to all DOE contractor facilities combined, not just WHC facilities. However, since the issuance of the current PFP FSAR, Westinghouse Hanford Company has revised, approved, and adopted their WHC-CM-7-5, *Environmental Compliance Manual* (May 15, 1992) which includes the most recent EPA requirements. Part D of this ECM specifically addresses the facility's current radioactive airborne emission controls and establishes a 10 mrem/yr EDE dose limit for offsite exposure.

Section 7.1.1.3.1, "Sampling Criteria for Radioactive Airborne Effluents," and section 7.1.1.3.2, "Monitoring Criteria for Radioactive Airborne Effluents," do not comply with sampling and monitoring requirements specified by NESHAPS regulations. Also, the monitoring guidelines in Section 7.2.1, "Radioactive Gaseous Effluents," do not reflect the current requirements. The NESHAPS regulations of 12/89 require sampling and monitoring for specific radionuclides in gaseous effluents from each facility that could by itself contribute (without effluent treatment systems in place) 1% of the EPA limit of 10 mrem/yr EDE, that is, 0.1 mrem/yr EDE. Again, WHC's WHC-CM-7-5, *Environmental Compliance Manual* (May 15, 1992) reflects the newer requirements. Part D of this ECM specifically addresses the facility's current radioactive airborne emission controls and requires sampling and monitoring to be performed consistent with the requirements in the above mentioned NESHAPS regulations.

Section 7.1.2.1, "Standards for Control of Radioactive Liquid Wastes," discusses discharge of liquids to the 216-Z-20 Crib. Discharging of radioactive liquids to the ground is to be discontinued in accordance with EPA-1089-03-04-120, *Hanford Federal Facility Agreement and Consent Order*, otherwise known as the Tri-Party Agreement (TPA). This document has been signed, and agreed to, by the EPA, DOE, and the Washington State Department of

Ecology. Discharges from PFP to the 216-Z-20 Crib were discontinued in 1994 in accordance with established TPA milestones.

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8.0 RADIATION PROTECTION

Chapter 8.0 of the PFP FSAR describes WHC's radiation protection program as it pertains to PFP. The radiation protection program is designed to limit radiation exposure of the work force and public, such that, radiation exposures are well below regulatory limits. The facility is operated so that radiation exposures are maintained within the regulatory limits stated in DOE Orders 5480.11, *Radiation Protection for Occupational Workers* and 5400.5, *Radiation Protection of the Public and the Environment*. Westinghouse Hanford Company has implemented the mandatory and recommended radiation protection standards as prescribed by DOE Order 5480.4, *Environmental Protection, Safety, and Health Protection Standards*.

8.1 OCCUPATIONAL RADIATION EXPOSURES

Discussion. Westinghouse Hanford Company's ALARA policy and organizational structure was specifically developed to incorporate the requirements of DOE Order 5480.11. The facility's ALARA policy was developed in accordance with DOE Order 5480.11. The radiation exposure of WHC employees, visitors, and members of the general public originating from facility operations is controlled to levels which are less than the applicable DOE limits. The following WHC manuals describe the organizational structure and the methods for controlling and ensuring that human exposures to hazardous substances and conditions are maintained to ALARA requirements: WHC-CM-1-1, *Management Policies*; WHC-CM-1-2, *Organization Charts and Charters*; WHC-CM-1-3, *Management Requirements and Procedures*; WHC-CM-4-13, *Operational Health Physics Procedures Manual*; WHC-CM-4-11, *ALARA Program Manual*.

Evaluation Findings. Westinghouse Hanford Company's ALARA program has been reviewed. The ALARA program was developed in accordance with the guidelines established in DOE Order 5480.11. Westinghouse Hanford Company's ALARA program has been implemented in accordance with DOE requirements and is therefore acceptable.

8.2 RADIATION SOURCES

Discussion. Westinghouse Hanford Company presents a description of the radiation sources located in the facility that were considered when the facility was designed. The facility was designed to process plutonium. The facility's radiation protection program assumes that any alpha contamination detected is plutonium and response actions are taken accordingly. This conservative assumption is applied to all operations at the facility. The radiation protection program is based on controlling alpha emitting radionuclides at or below those values established for ²³⁹Pu in accordance with DOE Order 5480.11. Good housekeeping practices are in place during operating campaigns and outages to minimize accumulation and provide cleanup of solids and liquids that may accumulate in process support or access gloveboxes. This is an ALARA practice that prevents a significant increase in personnel radiation exposures.

Two liquid effluent streams are generated by PFP operations. The radioactive liquid hood and laboratory wastes are collected in the sump tanks at the 241-Z Building for eventual routing to the tank farms. This stream is not discharged to the environment. Condensates, cooling water, and other "normally cold" (but potentially contaminated) drainage from PFP are routed to the 216-Z-20 Crib. Discharge to the 216-Z-20 Crib is to be discontinued in accordance with EPA-1089-03-04-120, *Hanford Federal Facility Agreement and Consent Order*, otherwise known as the *Tri-Party Agreement*. This document has been signed, and agreed to, by the EPA, DOE, and the Washington State Department of Ecology. Discharges from PFP to the 216-Z-20 Crib were discontinued in 1994 in accordance with established Tri-Party Agreement milestones.

The exhaust from process enclosures, which are sources of gaseous and airborne radioactive materials, is handled by the Zone 4 ventilation system. The Zone 3 ventilation system handles exhaust from areas in which there is a potential for the release of radioactive material. Zone 2 provides a buffer or transition area between Zone 3 and Zone 1. It is highly unlikely that Zone 2 areas would become contaminated or experience airborne radioactivity, but there is a low potential for materials to escape from Zone 3. Zone 1 contains offices, lunchrooms, etc., where there is little likelihood of contamination releases. This designation of ventilation control zones within the 234-5Z Building is reversed from the current definitions provided in DOE Order 6430.1A, *General Design Criteria*. The exhaust systems meet the requirements of DOE Order 6430.1A. Descriptions of anticipated off-normal occurrences are provided in SER Section 9.1.

Evaluation Findings. Facility operations involving the discharge of liquid effluent to the 216-Z-20 Crib have been discontinued in accordance with EPA-1089-03-04-120, *Tri-Party Agreement*. The exhaust systems used to control airborne radioactivity within the facility meet the requirements of DOE Order 6430.1A. The information presented in Section 8.2 of the PFP FSAR is acceptable.

8.3 RADIATION PROTECTION DESIGN FEATURES

Discussion. Radiation protection features were built into PFP. These features include a zoned HVAC system, compartmentalized process and analytical facilities, gloveboxes and hoods for hazardous operations, geometrically favorable SNM process and storage facilities, shielded storage, remote location of liquid lines for process and waste materials systems, shielded storage vaults and remote areas for surge, and accumulation tanks. Although the majority of the radiation protection design features were built prior to the development of DOE Order 6430.1A, the facility meets the intent of the requirements established for plutonium processing and handling facilities.

Personnel exposure to high levels of radiation and/or radioactive contamination is administratively and/or physically controlled. Radiation levels throughout the facility have been appropriately recorded and are based on readings taken during periods of plant processing operations. Radiation areas and high radiation areas are posted in accordance with

the requirements of DOE Order 5480.11. There are no "very high radiation areas" in the facility. Access to high radiation areas is controlled by a locked entrance to prevent unauthorized entry as required by DOE Order 5480.11.

Westinghouse Hanford Company limits occupational worker exposure to radiation in accordance with the administrative control limits established in the *Hanford Site Radiological Control Manual* (HSRCM). The HSRCM was released on December 1, 1992. The administrative control limits established in this manual supersede those presented in Table 8.3-2 (PFP FSAR pg. 8-145A). Table 8.1 provides the new WHC administrative control limits for occupational exposure and the annual limits established in the DOE N5480.6, *DOE Radiological Control Manual*. Westinghouse Hanford Company's established administrative control limits for whole body exposure are more restrictive than the 2.0 rem administrative control level prescribed in DOE N5480.6. Westinghouse Hanford Company's radiological control levels will allow them to meet DOE's objective to "maintain personnel radiation exposure well below regulatory limits."

Table 8.1. WHC and DOE Radiation Exposure Guidelines

Type of Exposure	WHC Annual Admin Control Limits (rem) ^a	WHC Annual Admin Control Limits (rem) ^b	WHC Annual Admin Control Limits (rem) ^c	DOE Annual Limits (rem)
Whole Body (internal + external)	0.5	0.75	1.0	5.0
Lens of Eye	4.5	6.75	9.0	15.0
Extremity				50.0
Extremity and Skin	15.0	22.5	30.0	
Any organ or tissue and skin				50.0
Any Organ	15.0	22.5	30.0	
Declared Pregnant Worker: Embryo/fetus	0.5 in 9 months	0.5 in 9 months	0.5 in 9 months	0.5 in 9 months
Minors and students: Whole body	0.1	0.1	0.1	0.1
Visitors and Public: Whole body	0.1	0.1	0.1	0.1

(a) Level 3 line manager & Radiological Controls Manager (RCM) approval required to exceed these limit.

(b) Level 2 line manager and RCM approval required to exceed these limits.

(c) Contractor senior site executive approval required to exceed these limits.

Fixed Head Samplers (FHSs) have been placed in strategic locations throughout PFP to evaluate the average airborne concentrations of contamination at work locations. The FHSs are located where there is a potential for airborne radioactive contamination concentrations to exceed 10% of the applicable Derived Air Concentrations (DAC) as required by DOE Order 5480.11.

Evaluation Findings. The PFP radiation protection design features meet the intent of DOE Order 6430.1A and are acceptable. Radiation areas throughout the facility have been established and posted in accordance with DOE Order 5480.11. Airborne contamination in the facility is monitored as prescribed by DOE Order 5480.11. Westinghouse Hanford Company has adopted the Hanford Site Radiological Control Manual (HSRCM) which is consistent with requirements in DOE N5480.6. The administrative control limits established in the HSRCM are more restrictive than DOE's radiation control levels and are acceptable. Westinghouse Hanford Company should update the radiological control limits in this chapter to be consistent with those limits established in the HSRCM during the FSARs next annual update. In general, WHC should ensure that the next annual FSAR update is done in such a way that all radiological controls information in the FSAR is consistent with the new information presented in the HSRCM.

8.4 ESTIMATED PERSON-REM PFP OPERATING PERSONNEL

Discussion. Westinghouse Hanford Company has establishes occupancy times for each room of the facility based on measured radiation fields obtained during plant operations. In the facility there are no areas with known airborne radioactivity concentrations that are easily accessible to, or normally occupied by, PFP workers. Areas with a high potential for airborne radioactivity are posted and locked as required by DOE Order 5480.11. Entrance to these areas requires special authorization as well as respiratory protection.

The major contributors of occupational exposure to radiation at PFP are associated with the operation of the RMC line, the plutonium recovery facility, and radiation protection technician support work. Personnel dosimetry data taken from 595 PFP workers in calendar year 1988 shows the cumulative exposure to be 144.51 person-rem, or an average of 243 mrem per person. Table 8.2 is taken from the PFP FSAR and provides a summary of the annual doses received by each functional work group in the facility during CY 1988. This summary is provided to demonstrate that the average worker exposure is well below the DOE established limits for radiation workers.

Evaluation Findings. Historical radiation exposure data shows that the average radiation worker will typically receive less than 10% of the 5 rem regulatory limit. This historical data provides evidence that the radiation protection program at PFP can be effective. The information presented in Section 8.4 of the PFP FSAR is acceptable.

Table 8.2. Summary of Annual Doses in PFP for CY 1988.

Functional Work Group	Average Whole Body (rem)	Total person-rem	Average Number of Personnel
PRF Operations	0.27	22.4	84
RMC Line Operations	0.45	46.8	105
Facility Services	0.08	4.6	60
Product Handling	0.41	16.5	40
Maintenance	0.19	17.8	96
Radiation Protection	0.40	24.6	61
Laboratory Personnel	0.08	4.04	53
PFP Engineering	0.08	7.67	96

8.5 HEALTH PHYSICS PROGRAM

Discussion. Section 8.5 of the PFP FSAR addresses the WHC Health Physics organization, training program, equipment, instrumentation, facilities, and procedures. Westinghouse Hanford Company's radiation protection program implements the requirements specified in DOE Orders by using various WHC control manuals. These documents are used by WHC to establish compliance with the requirements in DOE Orders 5480.4 and 5480.11.

The Health Physics (HP) program is a part of the Safety Department and Nuclear Safety Organization. The health physics management team is comprised of a PFP Health Physics manager, a PFP radiological engineer, a day supervisor, and four shift supervisors. The specific responsibilities and qualifications for each position have been well defined in the PFP FSAR.

The various HP-related portable equipment, instrumentation and facilities available to support the radiation protection program are described. Portable equipment (radiacs) and instrumentation (air sampler counters, CAMs, TLDs) meet the applicable ANSI standards required by DOE Order 5480.4. Table 8.5-3 in the PFP FSAR provides a summary of the minimum applicable ANSI instrumentation standards used at PFP for radiation detection instrumentation. Westinghouse Hanford Company uses thermoluminescent dosimeters (TLDs) instead of film badges to monitor personnel exposure to radiation. The TLD program performance criteria meet or exceed the intent of ANSI standard N13.15.

The Pacific Northwest Laboratories (PNL) provide the following radiation protection services for PFP and the Hanford site:

- Operation of the portable radiation detection instrument pool
- Radiation detection instrument calibration services
- The TLD services for personnel monitoring
- The HIEC chairmanship
- Environmental monitoring and sampling services
- Compilation and issuance of the annual Hanford Site Environmental Report
- Consulting services related to radiation and environmental protection.

Westinghouse Hanford Company's radiation monitoring methods, frequencies, and procedures are defined in WHC-CM-4-12, *Operational Health Physics Practices Manual* and WHC-CM-4-13, *Operational Health Physics Procedures Manual*. The facility has appropriate instruments available to monitor and measure dose rates for the purpose of controlling personnel exposure to radiation. The instruments are capable of measuring neutron, gamma, beta, and x-ray radiation fields as required by DOE Order 5480.11.

Both physical and administrative controls have been established to control access to, and stay times in, radiation areas. The facility access is controlled by the Hanford Patrol with entry restricted to those with proper credentials. Exterior doors, except the main entrance, are locked and/or alarmed. A receptionist is located at the main entrance to restrict entry to those having met the training required to enter a radiation area. As mentioned above, the interior doors to known airborne radioactivity areas are locked. Visitors are escorted while in radiation areas by plant personnel having an escort badge.

Radiation work permits are used to administratively control access to radiation areas in the facility. These permits provide guidance such as the radiological status of radiation areas, dosimetry requirements, and personnel protective equipment requirements for entry. Radiation areas are posted in accordance with the requirements in DOE Order 5480.11. Formal training programs are established to ensure that plant personnel receive adequate radiation safety and radiation worker training. The radiation protection training program for all employees, radiation workers, and health physics technicians is consistent with DOE Order 5480.11.

Westinghouse Hanford Company's radiation protection program maintains radiation training records, radiation program records, radiological status of work area records, ALARA records, and occupational radiation exposure records. These records are consistent with the requirements in DOE Order 5480.11. Specifically, the occupational radiation exposure records system is operated by PNL and the Hanford Environmental Health Facility (HEHF) in accordance with DOE Order 5480.11 and ANSI N13.6, *Occupational Radiation Exposure Records System*.

Evaluation Findings. Westinghouse Hanford Company's Health Physics program is consistent with the requirements established in DOE Orders 5480.11 and 5480.4. The DOE

review concludes that WHC's radiation training programs, radiation monitoring programs, radiac standards, and radiation records are acceptable.

8.6 ESTIMATED PERSON-REM OFFSITE DOSE ASSESSMENT

Discussion. Westinghouse Hanford Company assesses the contributions of PFP effluents to offsite exposure in the following manner:

- Identify the waste streams of concern
- Sample for specific radionuclides to determine routine releases
- Monitor for radioactivity to detect off-normal or transient levels of radioactivity that may exceed established limits
- Perform environmental monitoring of specific media to evaluate pathways of specific radionuclides to man.

Offsite environmental monitoring is provided by PNL under a DOE contract while onsite environmental monitoring is performed by WHC. Effluent sampling is conducted for those waste streams with a potential for exceeding 10% of the applicable DCG for airborne plutonium as prescribed by DOE Order 5400.5.

The effluent monitoring and sampling data is used to estimate the offsite doses. A conservative approach was used to estimate offsite doses by using the highest annual release data, and the worst case meteorological data for determining dispersion coefficients. The dose calculations take into consideration the various exposure pathways (i.e., inhalation of airborne contaminants, ingestion of waterborne contaminants, ingestion of airborne contaminants that have been deposited in foodstuffs, and direct exposure to airborne radionuclides).

Based on 1988 data for Hanford Site emissions, the maximally exposed individual (MI) offsite received 0.065 mrem EDE due to airborne radioactivity (100 times less than the 10 mrem EPA limit for airborne emissions). The maximum organ dose was 2 mrem received by the thyroid. The maximum organ dose associated with plutonium emissions was 0.026 mrem to the bone surfaces. And the total offsite dose for all exposure pathways was 0.08 mrem EDE. These numbers were based on total site emissions, which included contributions from PUREX and the 300 area facilities. The total offsite dose is well below the 100 mrem DOE limit established in DOE Order 5400.5.

Westinghouse Hanford Company references 25 mrem/yr EDE and 75 mrem/yr to any organ dose limits in Section 8.6.3.2, "Analysis of Effects and Consequences." These EPA radiological exposure limits only apply to sources from the management and storage of spent nuclear fuel, high-level, and transuranic wastes at disposal facilities. The EPA limit given in 40 CFR 61 as 10 mrem/yr EDE for all air paths, is a more appropriate EPA limit to use in place of the 25 mrem/yr and 75 mrem/yr limits referred to in this section. Since the issuance of the PFP FSAR, WHC has revised, approved, and adopted WHC-CM-7-5, *Environmental Compliance Manual* (dated May 15, 1992), which includes the most current EPA

requirements. Part D of this ECM specifically addresses the facility's current radioactive airborne emission controls and establishes a 10 mrem/yr EDE limit for offsite exposure. In general, all discussions throughout the PFP FSAR regarding offsite exposure limits have been superseded by the requirements established in WHC-CM-7-5, *Environmental Compliance Manual*.

The effluent and environmental monitoring requirements defined in Section 8.6.1, "Effluent and Environmental Monitoring" do not reflect the current requirements. DOE Order 5400.5 requires DOE facilities to comply fully with the requirements established in 40 CFR 61 *National Emissions Standards for Hazardous Air Pollutants* (NESHAPS) of 12/89. The NESHAPS regulations require the sampling and monitoring of specific radionuclides in gaseous effluents from each facility that could by itself contribute (without effluent treatment systems in place) 1% of the EPA limit of 10 mrem/yr EDE, that is, 0.1 mrem/yr EDE. This is not the requirement that is specified in Section 8.6.1. However, WHC's WHC-CM-7-5, *Environmental Compliance Manual*, reflects the current requirements. Part D of this ECM specifically addresses the facility's current radioactive airborne emission controls and requires sampling and monitoring to be performed consistent with the NESHAPS requirements.

In Section 8.6.3, *Estimated Exposures*, the paragraph beginning "The primary pathways of exposure to plutonium ..." refers to "cumulative" dose. This is no longer the correct terminology and should be calculated as EDE. Since 1988, DOE has adopted the effective dose equivalent (EDE). DOE defines the "committed EDE" as the 50-yr "committed" dose from inhalation and ingestion during the year of exposure. The EDE is defined as the sum of the committed EDE and the EDE from external exposure received in that same year (DOE/EH-0071, *Internal Dose Conversion Factors for Calculation of Dose to the Public*). Although the doses have not been determined in accordance with current DOE methodology, the cumulative doses are acceptable since they result in more conservative exposures estimates.

Evaluation Findings. The methodology used by WHC to estimate offsite doses was reviewed, as well as effluent sampling requirements, appropriate offsite dose limits, and the 1988 offsite dose estimates. The methodology used to estimate the offsite doses was conservative and is acceptable. The 1988 offsite dose resulting from all pathways was 0.08 mrem EDE and is well below the 100 mrem DOE limit established in DOE Order 5400.5. The offsite dose due to airborne radioactivity was estimated to be 0.065 mrem EDE, two orders of magnitude less than the 10 mrem EPA limit for airborne emissions. Based on the historical offsite dose estimates, WHC has proved that PFP can be operated well below the allowable radiological exposure limits established by DOE and the EPA. (Note: The dose estimates are based on data taken during actual plutonium processing operations, and therefore, present PFP operations will be less.)

The problems identified in this chapter are associated with information that has been superseded by other WHC documents (i.e. WHC's ECM) and have been clearly identified and documented in this SER. The deficiencies are not considered safety significant, and therefore,

the material as presented in Chapter 8.0, in conjunction with WHC's ECM, is suitable. Westinghouse Hanford Company should update information presented in Chapter 8.0 of the FSAR to be consistent with the information provided in their ECM during the FSAR's next annual update. In general, WHC should ensure that the next annual FSAR update is done in such a way that all environmental monitoring requirements in the FSAR are consistent with information presented in WHC's ECM.

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9.0 ACCIDENT SAFETY ANALYSIS

The DOE review and evaluation of the PFP FSAR accident safety analyses focused on the following considerations. In Section 9.0.1, the reviewers consider the evaluation criteria against which this chapter is evaluated. In Section 9.0.2, the reviewers consider the acceptability of WHC's definitions for both the Maximally Exposed Offsite Individual (MOI) and the Maximally Exposed Co-Located Worker (MCW)¹ along with WHC's proposed Risk Acceptance Guidelines (RAGs) and the criteria used to identify Safety Class Systems, Structures, and Components (SSCs) and select Operational Safety Requirements (OSRs). In Section 9.0.3, the reviewers consider the facility's hazardous material inventories and controls. In Section 9.0.4, the reviewers consider the adequacy of PFP's preliminary hazards analysis and the hazard categorization for the facility. Section 9.0.5 presents the methodology used by WHC to calculate radiological doses and Section 9.0.6 presents a list of worker safety programs at PFP.

An evaluation of each abnormal event and accident is presented in Sections 9.1 and 9.2. The goal of this review was to determine whether sufficient rigor and conservatism were used to determine the risk associated with facility operations and define the safety envelope, SSCs, and OSRs. A brief discussion regarding the uncertainties associated with the quantitative analyses is presented in Section 9.3 and a summary of the verification and validation of computer codes used by WHC to analyze the various accidents is presented in Section 9.4.

The DOE review of the safety analyses presented in the PFP FSAR (Draft 1991) began in June 1992. A list of 150 technical questions was prepared in July 1992 and sent by DOE to WHC requesting clarification and/or justification of specific information in the safety analyses. Westinghouse Hanford Company responded to the initial list of questions and by November 1992 all but nine technical issues remained to be resolved. The open issues were as follows:

1. Document the completion of all seismic upgrades required by WHC-SD-C084-RA-001 Rev. 1, *Plutonium Finishing Plant (PFP) Seismic Upgrade Review, Project C-084*. The safety analyses performed in the PFP FSAR (1991) assumes that the seismic upgrades referred to in WHC-SD-C084-RA-001 Rev. 1 are complete. Furthermore, this document specifically states that the analysis and upgrades contained within supersedes the upgrades required in Section 9.2.4a.1.2 of the PFP FSAR (Draft 1991).
2. Either a) document completion of the changes specified by WHC-SD-CP-SAR-012 Rev. 1, *MT-5 and HC-60 Safety Assessment* such as the seismically qualified MT-5 hot-plate power cut-off switch identified as a safety class 1 system or b) demonstrate that the radiological consequences are within acceptable limits. This switch is an

¹ Throughout the PFP FSAR and SER the terms Maximally Exposed Offsite Individual, Offsite Receptor, and Member of the Public are used interchangeably. Likewise, the terms Maximally Exposed Co-Located Worker, Onsite Receptor, and Onsite Worker have also been used interchangeably.

integral component of the seismic safety analyses and it is a safety class 1 system. The presence of this switch is credited for reducing the probability of a seismic-induced MT-5 explosion which was analyzed in the FSAR to have unacceptable consequences to an onsite receptor. If the switch is credited in the accident scenario then OSRs should be developed documenting surveillance requirements, limiting conditions of operation, and recovery procedures. If a re-analysis is performed to demonstrate that the consequences are acceptable then document the new analysis in the FSAR.

3. Dose consequences developed in the FSAR are based on a "best estimate" Pu inventory in the facility's gloveboxes, process lines, evaporators, etc. No controls exist to ensure that the facility operates inside this analyzed situation. Westinghouse Hanford Company must demonstrate how they intend to maintain the Pu inventory within its analyzed safety envelope. If WHC chooses to specify a new Pu inventory amount, other than that defined in the FSAR, then they should document the resulting radiological dose consequences (rem EDE) and demonstrate how they will maintain this new Pu inventory.
4. Demonstrate through detailed analysis that the frequency of a "red oil" explosion occurring in the various steam evaporators is below 1.0×10^{-6} , or that the consequences are below the risk guidelines presented in Section 9.4 of the FSAR. In addition, review the need for additional OSRs based on a more accurate analysis. Westinghouse Hanford Company needs to determine how they intend to include this work in the FSAR.
5. The 232-Z analysis as presented in the PFP FSAR results in unacceptable radiological dose consequences. It conservatively demonstrates that the resulting onsite radiological consequences are not acceptable (29.5 rem onsite) and states that a removal program exists to eliminate this problem. Since the removal program has not been undertaken, a more realistic analysis should be performed to determine the radiological dose consequences.
6. Change OSR 11.3.6 so that it reflects WHC's commitment to not use acetylene gas in PFP and also, to limit other flammable gases stored in the facility. This change reflects the fact that the deflagration analysis in the PFP FSAR is incorrect.
7. Delete the following OSR's:

SL 11.2.2, "Solution Temperature in Laboratory Concentrator"
LCO 11.3.7, "Hydrogen Fluoride Supply Seismic Shutoff Operability"
LCO 11.3.8, "Fluorinator Low Vacuum and Hydrogen Fluoride Shutoff Interlock"
LCO 11.3.9, "Hydrogen Fluoride Manifold High Pressure Shutoff Interlock"

LCO 11.3.10, "Filterbox 9AB Inline Continuous Air Monitor"
LCO 11.3.12, "Solution Temperature in Laboratory Concentrator"
DF 11.4.4, "Fluorinator Low Vacuum Manual Reset"
SL 11.2.1, "Criticality Prevention"

8. Demonstrate that a waste drum fire/explosion occurring on a loading dock outside of PFP is either incredible or that the consequences are acceptable. Waste drums are loaded with up to 400 g Pu and normally stored inside. A bounding fire/explosion accident inside of the facility was developed in the FSAR and found to be acceptable; however, the facility has not analyzed for, nor implemented controls to ensure that a waste drum fire/explosion outside of the facility is an incredible event.
9. Identify for each abnormal event and accident the OSRs and safety class systems used to mitigate the dose consequences to below the risk guidelines presented in Section 9.4 of the FSAR. Also, there exist additional concerns regarding the OSRs as presented in the PFP FSAR which have been provided to WHC. For example, OSR 11.4.3 should not be presented as a design feature rather it should be developed as a LCO, and OSR 11.3.3 should not invoke documents such as ANSI N509 and N510.

To resolve the open issues, several meetings were held between DOE, WHC, PNL, and INEL staff members and in March 1993, an agreement was reached with WHC on how to resolve the nine open issues. Due to the complexity of both the re-analyses and the technical review effort, the open issues were not completely resolved until May 1994, at which time a revision to Chapter 9.0 of the PFP FSAR was initiated.

Westinghouse Hanford Company documented the required re-analysis necessary to resolve the open issues in WHC Safety Assessment Documents (SADs). The information presented in the SADs was then incorporated into a major revision to Chapter 9.0 of the PFP FSAR. As the new analyses were developed, WHC decided to develop several supporting documents. Specifically, WHC-SD-CP-TI-190, *Technical Basis for Characterization of Plutonium for PFP Safety Analysis*, and WHC-SD-CP-TI-191, *Atmospheric Dispersion Coefficients for PFP Accident Analyses*, were developed by WHC during the re-analysis period and have subsequently been incorporated into the PFP FSAR (1994).

9.0.1 Evaluation Criteria

The evaluation criteria used by the DOE reviewers for evaluating the accident safety analyses are based on requirements and guidelines in the following DOE Orders, Standards, Policy Statements, and other references:

DOE Order 5400.5, Radiation Protection of the Public and the Environment, May 1990.

DOE Order 5480.5, Safety of Nuclear Facilities, September 1986.

DOE Order 5480.7, Fire Protection, November 1987.

DOE Order 5480.11, Radiation Protection for Occupational Workers, June 1992.

DOE Order 5481.1B, Safety Analysis and Review Systems, May 1989.

DOE Order 6430.1A, General Design Criteria, April 1989.

DOE Standard 1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports.

SEN-35-91, Nuclear Safety Policy, September 1991.

EPA-520/1-88-020, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. 1988.

NRC 1982, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. NUREG 1.145, Rev. 1, Nuclear Regulatory Commission, Washington D.C.

NRC 1979, Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing and Fuel Fabrication Plant. NUREG 3.35, Rev. 1, Nuclear Regulatory Commission, Washington D.C.

NRC 1990, MELCOR Accident Consequences Code System (MACCS). Model Description. NUREG/CR-4691, SAND86-1562, Vol. 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

UCRL 15910, Design and Evaluation Guidelines for DOE Facilities Subjected to Natural Phenomena Hazards. Lawrence Livermore National Laboratory Report, 1990.

HPS SDC 4.1, Hanford Plant Standard, Standard Design Criteria 4.1

Guidelines for Hazard Evaluation Procedures. Second Edition with Worked Examples, Center for Chemical Process Safety, 1992.

Hanna, S.R., G.A. Briggs, R.P. Hosker, Jr., 1982, Handbook on Atmospheric Diffusion. DOE/TIC-11223, National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia.

ANS/ANSI-8.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

9.0.2 Definitions, Risk Guidelines, and Selection Criteria

To determine the "risk" associated with an accident, a safety analysis should provide the consequences associated with an accident and an estimate of its annual probability of occurrence. The results of the analysis should naturally lead to the identification of SSCs and OSRs necessary to prevent or mitigate the accident. The consequences associated with the accident will normally be in terms of radiological dose or toxicological dosage. It is the relationship between the magnitude of the consequences and the annual probability of such an event that establishes the risk associated with a particular accident. This is consistent with the definition of risk:

Risk (R) is the relationship between the frequency (F) and the consequence (C) of an event such that,

$$R = C \times F.$$

A key parameter needed to determine, in a meaningful way, the radiological dose or toxicological dosage received by an individual is the location of the individual with regards to the point of release of a hazardous material. Westinghouse Hanford Company has established a method for determining the location for evaluating the dose or dosage received by both onsite and offsite receptors.

Once the consequence of an accident has been determined, the safety analyst must estimate the likelihood of such an event. By definition it is the relationship between consequence and likelihood that provides some measure of risk. To determine whether the risk is acceptable is a difficult task. Westinghouse Hanford Company has developed risk acceptance guidelines to judge the acceptability of the risk associated with a single accident.

From a practical stand point, the results of the safety analysis will enable a safety analyst to determine which systems, structures, and components provide preventative or mitigative features and what administrative controls might be needed to ensure that the facility operates within its analyzed safety envelope. Westinghouse Hanford Company has established selection criteria for identifying SSCs and criteria for developing OSRs.

Because the methodology used to 1) define receptor locations for establishing points of evaluation, 2) judge the acceptability of risk, 3) select SSCs, and 4) develop OSRs provides a foundation for developing the safety analysis, a brief evaluation of each is presented below.

9.0.2.1 Definitions of Offsite and Onsite Maximally Exposed Individuals

Discussion. Westinghouse Hanford Company presents definitions for offsite and onsite maximally exposed individuals (MI) in FSAR Section 9.3.3. Maximally exposed individuals are defined to establish locations for evaluating worst case exposure consequences for both the public and onsite worker. Westinghouse Hanford Company defines an offsite MI as a person

at the site boundary receiving the greatest radiological dose. The onsite MI is defined as a person located at the nearest normally occupied facility, not belonging to the PFP complex, and not included in the PFP facility emergency plan.

The specific locations were chosen by reviewing the atmospheric dispersion coefficients for each of the 16 compass sectors to determine the largest value with consideration given to the above definitions. The use of appropriate site specific meteorological data was verified. Differences between onsite and offsite receptor directions and distances are a result of the irregular site boundary, wind rose frequency, and meteorological conditions. The location, distance, and direction for each onsite and offsite receptor location is listed in Table 9.1. The differences in distance and direction between the elevated releases and the ground level releases result from the worst-case stability class, wind speed, wind rose frequency, and the credit taken for plume rise. The actual locations where dose and dosage consequences are evaluated for both the onsite and offsite receptors are presented in Figures 9.1 and 9.2.

Evaluation Findings. Westinghouse Hanford Company's definition of the offsite MI as being a maximally exposed individual located at the site boundary is found to be appropriate based on historic precedence. Part 100.11 of 10 CFR 100 requires that an exclusion zone be determined by calculating the distance from a reactor site at which an individual located at that distance would receive 25 rem as a result of a fission product release from the reactor. In that DOE considers the Hanford site to be similar in concept to the residential exclusion zone defined for nuclear reactors in 10 CFR 100, the DOE review concludes that defining the offsite MI as a maximally exposed individual located at the site boundary is appropriate.

The DOE review also concludes that WHC's selection of the onsite MI based on a hypothetical person located at the nearest normally occupied facility, not belonging to the PFP complex, and not included in the PFP facility emergency plan, is appropriate for evaluating dose and dosage for co-located workers.

Table 9.1 Locations of maximally exposed receptors for purposes of evaluating worst-case dose and dosage consequences

Release Scenario	Distance/Direction^(a)	Facility or Location where Receptor is Located
Onsite MI		
Elevated Release	930 m SSE of 291-Z-1 Stack	242-S
Ground Level Release	550 m WNW of 234-5Z building	272-WA
Buoyant Plume Rise	950 m ESE of PFP fence	U-Plant
Point Release at Ground Level	450 m WNW of PFP fence	272-WA
Point Release from 232-Z building	550 m WNW of 232-Z	272-WA
Offsite MI		
Elevated Release	12.5 km W of PFP	Western edge of Hanford Site Boundary
Ground Level Release	12.5 km W of PFP	(same)
Buoyant Plume Rise	17.4 km WNW of PFP	(same)
Point Release at Ground Level	12.5 km W of PFP	(same)

(a) The differences in receptor locations for the ground release scenarios and elevated release scenarios are attributed to the worst case meteorological conditions used. The receptor locations selected result in the smallest atmospheric dispersion, hence, the worst-case dose or dosage estimates.

Figure 9.1 Maximally Exposed Co-Located Worker Locations

Figure 9.2 Maximally Exposed Offsite Individual Locations

9.0.2.2 Radiological Risk Acceptance Guidelines

Discussion. To manage risk, it is appropriate to develop criteria for its acceptance. Westinghouse Hanford Company developed radiological Risk Acceptance Guidelines (RAGs) for judging the acceptability of radiological risk to both the public and the onsite worker. The RAGs are used to aid in the decision making process. The combination of risk guidelines and selection criteria for OSRs and SSCs allows the safety analyst to identify OSRs, ACs, and SSCs. As such, WHC's risk acceptance guidelines are simply guidelines and are not intended to be applied in an absolute manner for judging the acceptability of risk. Westinghouse Hanford Company's RAGs are based on nuclear safety goals established in SEN-35-91, *Nuclear Safety Policy*. For each of the nuclear safety goals a quantitative risk goal was defined. From the quantitative risk goals WHC developed risk acceptance guidelines.

Nuclear safety goals stated in SEN-35-91 address limiting risk to the general public from the operation of DOE facilities. Nuclear safety goal #1 limits the risk to an average individual in the vicinity of a DOE nuclear facility to one-tenth of one percent (0.1%) of the sum of prompt fatalities resulting from other accidents to which members of the public are generally exposed. Nuclear safety goal #2 limits the risk of cancer to an average individual in the vicinity of a DOE nuclear facility to one-tenth of one percent (0.1%) of the sum of all cancer fatality risks resulting from all other causes.

Westinghouse Hanford Company's RAGs for the **onsite** worker presented in Section 9.4 of the PFP FSAR are shown in Figure 9.3. The basis for WHC's RAGs are three different radiological dose values. A value of 1 rem EDE is assigned to an anticipated event (10^{-1} per year event frequency) and is derived from DOE Order 6430.1A Section 1300-6.2, *Shielding Design*. The 1 rem limit is based on a design requirement specifying that the shielding be designed to limit "the total EDE to less than 1 rem per year to workers." To apply the limit to colocated workers outside of the facility in the event of an accident is considered conservative. A value of 5 rem EDE is assigned to an unlikely event (10^{-2} per year event frequency) and is derived from DOE Order 5480.11. The 5 rem EDE limit is based on limiting the annual exposure (EDE) to a radiation worker to 5 rem. Its application to limiting the risk of onsite workers in the event of an unlikely accident is considered conservative. A value of 100 rem is assigned to an extremely unlikely event (10^{-6} per year event frequency) and is based on implementation of the second nuclear safety goal presented in SEN-35-91. A value of 100 rem EDE limits the risk of cancer fatalities (to a population in the area of PFP) resulting from a potential accident to less than one-tenth of one percent (0.1%) of the sum of all cancer fatality risks resulting from all other causes generally seen to occur in society.

Westinghouse Hanford Company's RAGs for the **offsite** worker also presented in Section 9.4 of the PFP FSAR are shown in Figure 9.4. A value of 0.01 rem EDE is assigned to an anticipated event (once per year) and is derived from DOE Order 5400.5. The order implements a regulatory requirement from the Clean Air Act thus reducing the allowable airborne emission of radionuclides from a DOE facility so as to limit the exposure to a

Figure 9.3. Onsite Radiological Risk Acceptance Guidelines (RAGs)

Figure 9.4. Offsite Radiological Risk Acceptance Guidelines (RAGs)

member of the public to less than 0.01 rem EDE per year. A value of 0.5 rem EDE is assigned to an unlikely event (10^{-2} per year event frequency) and is derived from DOE Order 5400.5. The order allows for a temporary increase in exposure of a member of the public to not greater than 0.5 rem in the event of an unusual circumstance at a DOE facility. To apply this value for judging the exposure resulting from an unlikely event to a member of the public is considered conservative. A value of 25 rem EDE is assigned to an extremely unlikely event (10^{-6} per year event frequency) and is derived from 10 CFR 100, *Reactor Site Criteria*. The criteria in 10 CFR 100 establishes reference values which can be used in evaluating reactor sites with respect to potential accidents of exceedingly low probability of occurrence. For this case it is considered both conservative and appropriate to assign a value of 25 rem to judge the acceptability of risk resulting from an accident with a one in one million probability. A 25 rem EDE value also meets the requirements established in the second nuclear safety goal.

Evaluation Findings. The DOE review concludes that WHC's radiological risk acceptance guidelines developed for both onsite and offsite receptors are more conservative than the nuclear safety goals established in SEN-35-91, and are therefore acceptable.

9.0.2.3 Toxicological Risk Acceptance Guidelines

Discussion. Emergency Response Planning Guidelines (ERPG) are guidelines for single short-term exposures that are of significance for emergency planning purposes. ERPG-1, ERPG-2, and ERPG-3 values are defined as follows:

ERPG-1: The maximum airborne concentration below which it is believed that nearly all members of the community could be exposed for up to one hour without experiencing other than mild, transient adverse effects such as slight irritation or a barely definable objectionable odor.

ERPG-2: The maximum airborne concentration below which it is believed that nearly all members of the community could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective action.

ERPG-3: The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

Westinghouse Hanford Company has developed onsite and offsite toxicological risk acceptance guidelines similar to their radiological risk guidelines (see FSAR Section 9.4.5). The offsite toxicological risk curve begins with an Emergency Response Planning Guide-1 (ERPG-1) value at a frequency of 1 and increases to a ERPG-2 value for accidents with a frequency of 1×10^{-6} . Westinghouse Hanford Company's onsite toxicological risk curve begins

with a ERPG-2 value for accidents occurring with a frequency of 1 and decreases to a ERPG-3 value for accidents with a frequency of 1×10^{-6} /year.

Evaluation Findings. The DOE has not developed specific criteria for determining the acceptability of risk resulting from accidental chemical releases. However, it is a common belief that the toxicological acceptance criteria used for individual hazardous chemicals should be similar to the concept of the Emergency Response Planning Guideline as developed by the American Industrial Hygiene Association (AIHA 1989). The DOE review concludes that the comparison of toxicological dosage estimates to ERPG values for the purposes of understanding the risk associated with a postulated accident is appropriate.

9.0.2.4 Criteria Used by WHC to Select Safety Class SSCs

Discussion. A facility's authorization basis will include those features of the facility's design that are considered to be important to operational safety and are relied upon by DOE to authorize the operation of the facility. Department of Energy Order 6430.1A requires a safety classification system and provides safety class criteria. The Order also specifies that the design of safety class systems, structures, and components must ensure that a single failure will not result in the loss of capability of a safety class system to perform its required safety function. To protect against single failures, safety class systems will be designed with sufficient redundancy and diversity to minimize the possibility of concurrent common-mode failures of redundant items.

In WHC-SD-CP-OSR-010, WHC establishes criteria for selecting Safety Class 1, 2, and 3 SSCs. (Note: WHC refers to Safety Class SSCs as Safety Envelope Design Features.)

Safety Class 1 and 2 SSC selection criteria:

1. Structural confinement boundaries necessary for prevention or mitigation of releases of radioactive materials outside of the facility,
2. Ventilation confinement boundaries necessary for prevention or mitigation of releases of radioactive materials outside of the facility,
3. Equipment features, such as that to prevent explosions, that prevent damage to structural or ventilation confinement or are used to maintain or monitor the integrity of the ventilation confinement boundary,
4. Equipment installed to provide an essential barrier utilized in the double contingency principle for prevention of nuclear criticality, and
5. Alarm systems used to evacuate personnel in the event of a nuclear criticality.

Safety Class 3 SSC selection criteria:

6. Internal confinement provided by established facility ventilation zones to prevent the spread of contamination within the facility; systems related to fire protection; and systems for explosion prevention. Failure of these systems could result in grossly jeopardizing facility worker safety.
7. Equipment used to warn personnel of airborne contamination in facility working areas, and
8. Equipment provided for safe evacuation of facility personnel in case of emergency.

In general, Safety Class 1 and 2 SSCs are relied upon in the safety analysis to prevent or mitigate adverse health effects to the public and co-located worker which may result from the release of hazardous material in the event of an accident. Safety Class 1 and 2 SSCs are subject to higher quality design, fabrication, and industrial test standards and codes and are controlled by a QA program consistent with NQA-1. Safety Class 3 SSCs protect the facility workers by ensuring confinement of hazards materials, and in the event of a release, notification of an abnormal condition which may warrant evacuation from the facility. Safety Class 3 SSCs are subject to standard design, fabrication, and industrial test standards and codes.

Administrative Control 5.23, *OSR Compliance Program* specifies that a program be implemented to ensure compliance with the OSRs. A key element of this program is the definition and functionality of plant SSCs. Accordingly, WHC has developed a series of System Description Documents that provide detailed system descriptions, operability definitions, and functional requirements necessary for maintaining the SSCs.

Evaluation Findings. DOE Order 6430.1A defines safety class items as those systems, components, and structures with the following characteristics:

- Those whose failure would produce exposure consequences that would exceed the guidelines in Section 1300-1.4, *Guidance on Limiting Exposure of the Public*, at the site boundary or nearest point of public access
- Those required to maintain operating parameters within the safety limits specified in the OSRs during normal operations and anticipated operational occurrences
- Those required for nuclear criticality safety
- Those required to monitor the release of radioactive materials to the environment during and after a Design Basis Accident (DBA)
- Those required to achieve and maintain the facility in a safe shutdown condition

- Those that control the safety class items described above.

Westinghouse Hanford Company has established a set of criteria to identify SSCs and has developed a series of documents to provide system descriptions, operability definitions, and functional requirements to maintain them. A list of PFP Safety Class 1 and 2 SSCs are presented in Table 4.3. The DOE review concludes that the criteria established by WHC to classify SSCs is consistent with those requirements set forth in DOE Order 6430.1A and is therefore acceptable.

9.0.2.5 Criteria Used by WHC to Select Operational Safety Requirements

Discussion. A facility's authorization basis will also include those requirements that define the conditions, safe boundaries, and management or administrative controls necessary to ensure the safe operation of a nuclear facility and to reduce the potential risk to the public, co-located worker, and facility worker from an uncontrolled release of radioactive materials or from radiation exposure due to inadvertent criticality. Operational Safety Requirements define the conditions, safe boundaries, and management/administrative controls necessary to ensure the safe operation of a nuclear facility.

The initial FSAR review effort concluded that the original OSRs proposed in Chapter 11.0 of the PFP FSAR (1991 Draft) were inadequate. In general, the OSRs were significantly lacking in technical rigor, structure, format, and content. As a result, WHC made significant improvements in the facility's OSRs. Westinghouse Hanford Company's currently approved OSRs are documented in WHC-SD-CP-OSR-010, *Plutonium Finishing Plant Operational Safety Requirements*.

Westinghouse Hanford Company selected OSRs for the facility based on defining the facility's operating conditions, safe boundaries, and management and administrative controls necessary to prevent or mitigate the adverse consequences of an accident on the health of the public, co-located worker, or facility workers. Specifically, selection criteria were based on numerical threshold radiological and toxicological dose criteria in FSAR Section 9.2.4 and DOE Order 5480.22 and its predecessors.

Evaluation Findings. Westinghouse Hanford Company's criteria for selecting OSRs is consistent with the requirements set forth in DOE Orders 5480.5 and 5480.22. The DOE review concludes that WHC's OSR selection criteria is appropriate.

9.0.3 Hazardous Material Inventories and Controls

Discussion. An approximate inventory of non-radioactive hazardous materials is presented in Chapter 6.0, Section 6.1.3.2 of the PFP FSAR. Chemical related accidents and hazards in the facility have the potential to cause damage to equipment and can result in worker injury. Accident safety analyses of postulated worst case toxic release scenarios were developed by WHC and included in the FSAR. These scenarios are evaluated in SER Sections 9.2.4A.13,

Failure of the HNO₃ Storage Tank, 9.2.4A.14, Seismically-Induced Spill of CCL₄, 9.2.5.1, HNO₃ Storage Tank Rupture, 9.2.5.2, CCl₄ Drum Rupture, and 9.2.8, Toxic Chemical Release (CCL₄).

The review effort concentrated on the plutonium inventory within the facility. Initial findings concluded that the dose consequences developed in the FSAR, and in particular the seismic analyses, were based on a "best estimate" plutonium inventory within the facility's gloveboxes, fume hoods, process lines, and evaporators and that inadequate controls were in place to ensure that facility operations were conducted within the assumed inventory in the analysis. In essence, this meant that there was no means to maintain facility operations within the analyzed safety envelope.

As a result of the initial findings, WHC re-evaluated their current plutonium inventory and controls, and re-analyzed the radiological consequences associated with the seismic event. Westinghouse Hanford Company chose to rely on its criticality prevention program to provide adequate inventory controls. In most cases, the facility's Criticality Prevention Specification (CPS) Limits for each glovebox, fume hood, and process flowsheet were used to describe the inventory of plutonium assumed within the facility for the purpose of performing accident analyses. By using the facility's CPS limits to represent the plutonium inventory analyzed in the safety analyses, WHC is able to ensure that the facility will indeed operate within its analyzed safety envelope. Furthermore, the use of CPS limits to describe the facility's inventory for purposes of analyzing accidents provides an additional level of conservatism in the analysis.

Westinghouse Hanford Company developed LCO 3.3.1, *Control of DISPERSIBLE Plutonium*, to control the facility's plutonium inventory in the various laboratories, gloveboxes, and process areas.

Evaluation Findings. From a review of the chemical inventories located at the facility and given the distance from the facility to the "public," the DOE review concludes that the toxicological risks to the public are minimal. However, accidents involving hazardous chemical materials have been developed by WHC for inclusion in the PFP FSAR.

The DOE review concludes that there are sufficient controls to ensure that PFP plutonium handling operations will be conducted in such a manner that the actual facility inventory will not exceed the plutonium inventory analyzed in the various accidents. The DOE review also concludes that by developing a theoretical plutonium inventory based on CPS limits, that WHC has provided an additional level of conservatism with regards to developing source term values for the various accident scenarios.

9.0.4 Preliminary Hazards Analysis Evaluation and Hazard Categorization

Discussion. The events selected for analysis by WHC were determined from situations in which combinations of energy and toxic materials exist in significant quantity to potentially

create unacceptable consequences as depicted in the logic tree of Figure 9.0.2-1 of the PFP FSAR (WHC 1994). The top event for the logic tree is an atmospheric release of radioactive material which is determined from five lower levels of events. The lower level events are evaluated as either an abnormal event or an accident depending primarily on the estimated quantity of material released and the frequency of the event occurring. Events were selected from 1) a review of plant history to identify events that have occurred, 2) a walkthrough of the facility to identify potential hazards, and 3) a systematic search to discover any events that may have been missed. Hypothetical worst case events were determined for each category of hazard. These worst case hazard events were evaluated to assess onsite and offsite consequences, and further analyses were performed for cases in which the consequences were unacceptable. Only those events that were determined from the hazard analyses to have a probability of occurrence greater than $1.0 \times 10^{-6}/\text{yr}$ and cause some adverse consequence were included in the PFP FSAR Chapter 9.0. Events such as loss of power, hazards from nearby facilities, and other offsite hazards were not analyzed in detail because they either resulted in a probability of occurrence less than 1.0×10^{-6} or they had no adverse consequence. Included in the hazard analysis were the impact of natural forces to the PFP. The analysis considered flooding, strong wind, wind generated missiles, lightning, snow fall, earthquakes, and volcanic ash loading. The FSAR addressed characteristics associated with each of these natural phenomena (see Section 4.1 of the SER) that could affect the facility and WHC correctly concluded that only the design basis earthquake and strong wind scenarios had any potential for significant onsite and/or offsite consequences.

After reviewing the various accidents identified by WHC in the PFP FSAR, an issue was raised regarding the potential of a fire/explosion occurring in a 55 gallon drum containing radioactive waste stored outside the facility. This accident scenario had been identified at other DOE sites and is a plausible scenario at PFP. Westinghouse Hanford Company resolved this issue by analyzing the accident scenario and including it in the PFP FSAR.

Evaluation Findings. Westinghouse Hanford Company reviewed the plant's 45 years of operating history to identify any previously documented safety significant events and then conducted facility walkthroughs to further identify any additional potential hazards. Techniques employed by WHC to perform the facility's preliminary hazard assessment are similar to those presented in *Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples* (1992). The DOE review concludes that WHC has performed an adequate preliminary hazards assessment to identify the facility's potential accidents.

Although the PFP FSAR was issued prior to DOE Order 5480.23 and DOE-STD-1027-92, DOE believed it was appropriate to assign the facility a Hazard Category in accordance with the most up to date standard. Based on WHC's preliminary assessment of hazards and the facility's radioactive material inventory, the DOE review concludes that the Plutonium Finishing Plant is a **Hazard Category 2** facility per DOE-STD-1027-92. This means that the hazard analyses show the potential for significant on-site consequences.

9.0.5 Radiological Dose Calculations

Discussion. Radiological dose calculations used in the various accident safety analyses were performed by WHC using the following equation:

$$D(\text{rem}) = Q(\text{g}) \times \overline{\chi}/Q'(\text{sec}/\text{m}^3) \times R(\text{m}^3/\text{s}) \times C(\text{rem}/\text{g})$$

where,

D = Effective Dose Equivalent (EDE) (rem)

Q = quantity of respirable plutonium released into the atmosphere² (grams)

$\overline{\chi}/Q'$ = atmospheric dispersion coefficient (sec/m³) location dependent

R = standard man breathing rate (m³/s)

C = Unit Inhalation Dose Conversion Factor (rem EDE/gram)

Westinghouse Hanford Company's derivation of atmospheric dispersion coefficients and unit inhalation dose conversion factors are evaluated in the following sections. The "standard man breathing rate" is a constant and is defined by the International Commission on Radiation Protection (ICRP 23). The evaluation of the quantity of respirable plutonium released into the atmosphere is unique to each individual accident scenario and is therefore deferred to the various accident evaluation sections.

Westinghouse Hanford Company did not consider the contribution to total dose resulting from air submersion, ingestion, or exposure to ground contamination. Isotopes of plutonium and americium are strong alpha emitters and, therefore, exposure due to inhalation is the most significant contributor to total dose. Also, sufficient conservatism exists in the overall dose calculation in terms of specifying source terms, respirable release fractions, leakpath factors, etc., to account for the small percent contribution to dose resulting from pathways other than inhalation.

Evaluation Findings. The method used by WHC to perform radiological dose calculations is technically correct and therefore acceptable. The method's technical justification can be found in a variety of Nuclear Engineering and/or Health Physics textbooks and NRC regulatory guides. The DOE reviewers conclude that sufficient conservatism exists in the overall dose calculation in terms of specifying source terms, respirable release fractions, leakpath factors, etc., to account for the small percent contribution to dose resulting from pathways other than inhalation.

² Q may be divided into the following five factors: Material at Risk, Damage Ratio, Airborne Release Fraction, Respirable Fraction, and Leakpath Factor.

9.0.5.1 Evaluation of Atmospheric Dispersion Coefficients

Discussion. Atmospheric dispersion coefficients ($\overline{\chi}/Q'$) represent the dilution of an airborne contaminant due to atmospheric mixing and turbulence. To facilitate atmospheric dispersion coefficients calculations, WHC developed a FORTRAN code referred to as GXQ (refer to PFP FSAR Section 9.3.2). The DOE review of WHC's $\overline{\chi}/Q'$ values focused mainly on the technical adequacy of the atmospheric dispersion models used in the GXQ code and whether the $\overline{\chi}/Q'$ values generated were reasonable. In addition, comparison calculations were performed using both GENII and MACCS computer codes³.

Atmospheric conditions do fluctuate, therefore a conservative atmospheric condition can be defined as a condition which causes a downwind concentration of airborne contaminants to be exceeded only 0.5% of the time (NRC 1982). The GXQ code uses 9-year averaged Hanford site specific joint frequency meteorology data to calculate 99.5% worst case $\overline{\chi}/Q'$ values. Westinghouse Hanford Company developed $\overline{\chi}/Q'$ values to describe four separate accident scenarios.

- *Elevated Release* $\overline{\chi}/Q'$ values to represent the atmospheric dispersion resulting from a stack release.
- *Ground Level Release with Building Wake and Plume Meander* $\overline{\chi}/Q'$ values to represent the atmospheric dispersion resulting from the release of material in the "no-ventilation" seismic scenario.
- *Buoyant Plume Rise* $\overline{\chi}/Q'$ values to represent the atmospheric dispersion resulting from a postulated ground level fire in stored waste drums located outside of the facility.
- *Point Source Ground Level Release* $\overline{\chi}/Q'$ values to represent the atmospheric dispersion resulting from a point release located outside the facility.

³ The GENII code (Napier 1988) was developed at Hanford to calculate radiation doses received by individuals and population groups as a result of chronic and acute gaseous and liquid effluent releases to the environment. The GENII code has been recommended for use at Hanford by the Hanford Environmental Dose Overview Panel (HEDOP) (Rhoads 1992) and DOE/RL. The GENII code has received extensive peer review from PNL staff members and two review panels of experts in environmental dosimetry external to PNL. Furthermore, the code has been reviewed by Argonne National Laboratory personnel and adopted for use in the New Production Reactor (Stull 1990). It has been formally recommended by the U.S. Nuclear Regulatory Commission and Sandia National Laboratory for use in low-level waste performance assessments (NRC 1989), and has also been adopted by the Environmental Institute of the Joint Research Center of the Commission of the European Communities. MACCS is a multi-featured nuclear accident consequences assessment code developed by Sandia National Laboratory with funding from the USNRC. Idaho National Engineering Laboratory performed the codes independent verification and documented that effort in the report NUREG/CR-5376, *Quality Assurance and Verification of the MACCS Code*.

Evaluation Findings. Atmospheric dispersion calculations were performed using the recently developed WHC code referred to as GXQ. The code computes atmospheric dispersion coefficients using a basic Gaussian straight line plume model. The code also allows for various corrections to be made depending on its specific application (i.e., building wake, plume meander, stack downwash, etc.). The gaussian straight line model is per NUREG 1.145 and the building wake model is per NUREG/CR-4691. The plume meander model is a 5th power law plume model taken from Hanna (1982). The DOE review also verified that appropriate Hanford site-specific joint frequency meteorology data were being used.

The dispersion coefficients produced by GXQ for the *Point Source Ground Level Release* scenario appear to have a reasonable level of conservatism. Numbers generated by GXQ tend to be similar to those generated by GENII, with differences between model results accounted for by differences in their respective computational methods.

The dispersion coefficients produced by GXQ for the *Ground Level Release with Building Wake and Plume Meander* release scenario appear to have a reasonable level of conservatism. Numbers generated by GXQ tend to be similar to those generated by GENII, with differences between model results accounted for by differences in their respective assumptions and computational methods. The GXQ building wake model appears to give more conservative results than the GENII building wake model.

The dispersion coefficients produced by GXQ for both the *Elevated Release* and *Buoyant Plume Rise* scenarios were initially determined using a plume rise model that was technically inadequate, based on the original references that the model was derived from and, therefore, not conservative. This issue was brought to WHCs attention and as a result a limit was placed on plume rise to ensure that the code would conservatively estimate plume height. The revised elevated plume height $\overline{\chi}/Q'$ values compared well with similar values determined using the MACCS code.

The DOE review concludes that WHC's atmospheric dispersion coefficients are acceptable for use in the PFP safety analyses. However, the specific application of each $\overline{\chi}/Q'$ value in the various accident safety analyses is evaluated on a case-by-case basis in the DOE review of individual accident scenarios.

The technical acceptance of the atmospheric dispersion coefficients does not constitute formal acceptance of the GXQ code by DOE. The code may be misapplied if incorrect models are chosen for a specific application. For example, the user manual does not appear to clearly define which model should be applied to different applications. Until the users manual for the code is reviewed and approved, each application of this code needs to be reviewed. Future uses of the code by WHC to perform safety analysis shall require separate independent evaluations until the code is formally accepted by DOE.

9.0.5.2 Evaluation of Unit Inhalation Dose Conversion Factors

Discussion. Radiological exposure calculations performed in the PFP FSAR (1991 Draft) used unit inhalation dose values based on 5 year old 6% Pu-240. This meant that the source term was assumed to be 5 year old plutonium containing less than or equal to 6% Pu-240. During the FSAR review effort a concern was raised by both DOE and internal Westinghouse reviewers regarding the conservatism associated with assuming the source term was 6% Pu-240 material.

Initial plutonium isotopic mixture is determined at the time of production. Processing activities after production remove americium from the plutonium; however, after the initial production and processing phases Pu-241 beta decays to produce americium-241. As the plutonium ages, the americium content will slowly increase until a certain time at which point the americium content will begin to decrease. The alpha decay of Am-241 is a significant contributor to the total unit inhalation dose and therefore must be considered. Also, as the material ages the amount of Am-241 increases and its contribution to the total unit inhalation dose value will increase.

As a result of the concerns raised regarding the isotopic mixture of plutonium assumed in the source term, WHC re-analyzed the characteristics of their source term and presented revised unit inhalation dose values in WHC-SD-CP-TI-190, *Technical Basis For Characterization of Plutonium for PFP Safety Analyses*. PFP FSAR (WHC 1994) Section 9.3.5.2, *Pu Inventory Characterization* summarizes the results of WHC-SD-CP-TI-190 and provides the unit inhalation dose values which have been incorporated into the accident analyses.

The revised unit inhalation dose values represent the actual inventory of plutonium stored within the facility and conservatively account for plutonium aging effects. Westinghouse Hanford Company has divided their inventory into two categories. The first category represents plutonium with greater than 10% Pu-240 content, and the second category represents plutonium with less than 10% Pu-240. Mass weighted minimum, maximum, average, and 90th percentile unit inhalation dose values have been developed for each category. The values were corrected to account for worst case aging effects.

In most cases WHC used the 90th percentile worst case unit inhalation dose to perform accident dose calculations. Where WHC used average values, sufficient justification was required by the reviewers.

Evaluation Findings. A detailed review of the methodology used by WHC to develop their unit dose inhalation values in WHC-SD-CP-TI-190 was performed. The DOE review verified that the correct dose conversion factors were used for the various solubility classes per EPA-520/1-88-020. The DOE review concludes that WHC's methodology for developing unit inhalation dose values is both technically justifiable and conservative. However, the use of these values in the various accident scenarios must be evaluated on a case-by-case basis in the

DOE review of individual accident scenarios. In general, the methodology used to develop the unit dose inhalation values is acceptable.

9.0.6 Worker Safety Programs

Discussion. The accident safety analyses documented in Chapter 9.0 do not address, for the most part, worker safety issues. Rather, safety assurance for the workers is provided by establishing minimum operability requirements for criticality safety systems, by identifying Safety Class 3 SSCs, and by implementing and maintaining a variety of institutional safety programs at the facility. Westinghouse Hanford Company's WHC-SD-CP-OSR-010 document also presents Administrative Controls (ACs) which address several worker safety programs. Other specific institutional programs which provide for worker safety at PFP are:

Conduct of Operations	Emergency Preparedness
Training and Qualification	Quality Assurance
Criticality Protection	Radiation Protection
Fire Protection	Radiation Monitoring
Maintenance Management	Unreviewed Safety Questions
Operational Readiness Review	

Evaluation Findings: Westinghouse Hanford Company's Conduct of Operations, Training and Qualification, and Emergency Preparedness programs are discussed in Chapter 10. Their Quality Assurance program is discussed in Chapter 12. Radiation protection and Monitoring are briefly addressed in Chapter 8.0, and their Criticality protection program is considered in the criticality accident evaluation.

9.1 ABNORMAL OPERATIONS

Abnormal operational occurrences, as opposed to accidents, are those transients resulting from equipment failures or operator errors that are expected to occur during normal or planned modes of facility operation. Separation of events for inclusion as abnormal operations was based on both probability of the event occurring and on the apparent impact on plant personnel. These abnormal operational events are assumed to occur with an annual probability of 1.0

Abnormal operation scenarios are generic and were developed from a compilation of events recorded over 40 years of PFP operations, and include additional hypothetical upset conditions. A review of the plutonium handling experiences for the 15 year period 1972 to 1986 identified 34 unusual occurrence reports that documented abnormal events resulting in, or with the potential for, an onsite release hazard. Annual occupational exposure records were also reviewed. Events that require more than two independent errors or failures were generally excluded from evaluation as an abnormal event based on a low probability of occurrence. However, in a few instances, large impact events that involve more than two independent failures were evaluated as bounding case scenarios.

The transient events included in the Abnormal Operations Section were categorized in terms of the following system variations:

- (1) Radioisotope-Bearing Liquid Spills - The transient analyzed spilling a 10 liter bottle containing 16.4 kg of plutonium nitrate solution. The amount of Pu dispersed during this event bounds all possible abnormal radioisotope-bearing liquid spills and is based on the maximum inventory allowed by the Criticality Prevention Specification limits.
- (2) Radioisotope-Bearing Pressurized Liquid Container - The transient analyzed overpressurizing a 10 liter bottle containing 16.4 kg of plutonium nitrate solution. The amount of Pu dispersed during this event bounds all possible abnormal radioisotope-bearing liquid containers that may become pressurized and is based on the maximum inventory allowed by the Criticality Prevention Specification limits.
- (3) Radioisotope-Bearing Powder Spills - The transient analyzed spilling a slip-lid can containing 1 kg of plutonium oxide powder. The amount of Pu dispersed during this event bounds all possible abnormal radioisotope-bearing powder spills.
- (4) Radioisotope-Bearing Pressurized Powder Container - The transient analyzed overpressurizing a slip-lid can containing 1 kg of plutonium oxide powder. The amount of Pu dispersed during this event bounds all possible abnormal radioisotope-bearing powder container pressurization.
- (5) Stack Release Resulting from Filter Changes - Transients analyzed in this group included a dropped filter (containing 0.9 g of Pu) in filter room 309 or 310 and a dropped filter in filter room 26 in the 236-Z Building.
- (6) Overpressure of Remote Mechanical C (RMC) Line Pressure Vessel - The RMC line pressure vessel as analyzed in the FSAR is used for producing plutonium metal. Since the RMC line will now produce only plutonium oxide, this transient was not reviewed as part of the SER.
- (7) Ventilation System Upset - Transients analyzed in this group included excessive pressurization and depressurization of the following buildings: 234-5Z, 236-Z, 2736-ZB, 2736-Z, and 241-Z.
- (8) Loss of Ventilation - The transient analyzed was a complete loss of ventilation air flow in the 234-5Z, 236-Z, and 242-Z Buildings.
- (9) Glovebox Breach - The transient analyzed was a glove box penetration and subsequent release of material. This scenario was qualitatively developed and shown to be bounded by other, more limiting, abnormal events.

9.1.1 Radioisotope-Bearing Liquid Spills

Discussion. This event is forwarded from preliminary analysis (PA) elements B-2 and E-1 (see PFP FSAR Section 9.0.2.4). Due to the frequent handling of contaminated liquids, the probability of a liquid spill is very high. In the past, spills have been caused by improper routing of glovebox connections, accidental discharge of sprinkler systems in contaminated areas, failed seals, leaking valves, overfilling of containers, unanticipated backflow, operator error, management error, and corrosion of protective barriers. The composition of the spills can run from slightly contaminated waste water to liquids containing significant amounts of radionuclides.

Virtually any liquid handled at PFP has some probability of being spilled. These spills can occur anywhere within the facility that the liquids are handled. However, the PFP design, equipment, systems, training, management, and operations are intended to reduce the likelihood of large spills with concentrated material. All radioisotope-bearing liquids handled in the PFP are confined within systems such that a single failure will not result in the release to the environment. Contaminated liquids are handled within enclosed gloveboxes or hoods that have preferential airflow away from operational personnel. Liquids are not brought out from confinement until they have been enclosed in several protective layers which include containers, canisters, and sealed plastic bags.

The PFP has at least three redundant confinement systems that protect against an inadvertent release of radionuclides. These confinement systems include:

- Primary Confinement System - Consists of the process enclosures and their associated ventilation system. The process vessels within the enclosures (i.e., cans and canisters) can also be considered as part of the primary confinement system.
- Secondary Confinement System - Consists of the barriers that enclose the areas (rooms) that house the primary confinement system and the systems that ventilates those areas.
- Tertiary Confinement System - Consists of the building structure and its ventilation system; composed of confinement barriers (i.e. the building walls) and the ventilation system.

For the purposes of the analysis, it was assumed that the entire contents of a 10-L polyethylene or steel bottle containing 16.4 kg of plutonium nitrate solution was spilled outside of a glovebox located within the facility from a height of 3 m. The 10-L container of plutonium nitrate solution represents the largest total quantity and most concentrated solution that is routinely handled outside the process confinement in proximity to facility staff. The analysis is performed in a conservative manner by assuming that the spill height is twice the maximum lift height of 1.5 m. It is also conservatively assumed that the 10-L polyethylene bottles are still used in this task when in fact they have been replaced with steel containers.

The steel containers now used have passed drop tests without leaking. All other protective systems are assumed to be operating normally prior to the spill. The consequences of this event bounds all other radioisotope bearing liquid spill events credible in the PFP.

As a result of a spill from a height of 3 m, it was estimated that 0.002% by weight of the plutonium nitrate solution will become airborne as droplets using NRC approved methodology presented in NUREG/CR-2139, *Aerosols Generated by Free Fall Spills of Powders and Solutions in Static Air*. This results in 0.16 g of Pu becoming airborne. Upon detection of the spill, the spill area is evacuated and confined to people with breathing apparatus for cleanup, limiting the exposure to facility personnel. Any airborne particulates are discharged into the E-3 or E-4 system from Zones 3 or 4 with filtration through at least a single stage roughing filter and a single stage of HEPA filtration. The HEPA filters have an efficiency of 99.95% resulting in less than 8.0×10^{-5} g of Pu reaching the PFP stack.

Evaluation Findings. The possibilities for spills of contaminated liquid have been reviewed. Several initiating events can lead to a liquid spill. WHC determined that the bounding liquid spill accident in regard to radiological consequences was the spill of a 10-L container of plutonium nitrate solution. The scope of this review has included investigations of initial conditions, assumptions, and results of the liquid spill. The method used to determine the extent of airborne dispersal has also been examined. The worst case Pu release in the room due to the spill is 0.16 g. Using one stage of HEPA filtration, the worst case Pu release is 8.0×10^{-5} g. The HEPA efficiency is maintained by OSR 3.2.1, *Filtered Exhaust From Zones 3 and 4*, which specifies that the minimum acceptable efficiency for filtration of radioactive particles from the ventilation exhaust stream is 99.95 and specifies the minimum number of HEPA filter stages in the ventilation exhaust stream.

The release of 8.0×10^{-5} g results in 2.8×10^{-5} rem EDE onsite and 1.0×10^{-5} rem EDE offsite dose consequence. These results are well below the DOE requirements and the risk guidelines presented in Section 9.4 for an annual event and, are therefore, acceptable. The basis for acceptance in the review is that WHC's choice of the maximum liquid spill accident has been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.2 Radioisotope-Bearing Pressurized Liquid Container

Discussion. This event is forwarded from PA element D-19 which identifies events that may lead to the pressurized discharge of radioisotope-bearing liquid into occupied areas (see PFP FSAR Section 9.0.2.4). Liquid containers become pressurized due to chemical decomposition of the contained liquids, irradiation-induced degradation of the container material, chemical corrosion of the container material, or radiolytic gas buildup. Pressurization can also result from the application of heat and container mishandling.

Plutonium nitrate solution is stored in 10-L containers, which represents the largest total quantity and most concentrated solution that is routinely handled outside a process enclosure

in proximity to the plant staff. Laboratory sample sized containers and dilute quantities from cleanup operations are also handled outside of confinement, but they involve smaller quantities of hazardous material and are bounded by the consequences of pressurization of the 10-L container.

Essentially two situations can occur to cause an overpressurized container to result in an abnormal event. The first occurs when, upon opening the overpack, the operator observes that the primary container has failed due to pressurization. The second is when the pressure is not relieved from the liquid container because of either a plugged vent valve or improperly applying the venting procedure. Contaminated liquids are handled within enclosed gloveboxes or hoods that have preferential airflow away from operational personnel, and they are not brought out of confinement until they have been enclosed in a number of different layers of protective containers and canisters. All liquids are first placed into a primary container and then that container is placed into an overpack FL-10, L-10, or 10-L container. Pressurization of a liquid container does not constitute an abnormal event until it causes failure of its primary container or causes an event when an attempt is made to transfer the liquid. The result of the primary container failure would be mitigated by the secondary container (overpack).

Due to physical and procedural controls, the potential for significant discharge of a pressurized container into an occupied area is low. Plutonium solutions are currently stored only inside the PFP and are delivered to the 227 room hood as needed for repackaging. Due to a prohibition against shipping radioactive solutions offsite, it is anticipated that the PFP will not receive any additional offsite plutonium solutions. Shipping containers for liquids have special provisions for venting internal pressure, while storage containers have vents built in that allow continual venting of the interior.

For the purpose of this analysis, WHC assumed that the liquid containers are handled in a normal manner and operators are wearing supplied air respirators. When a 10-L container containing 16.4 kg of plutonium nitrate solution with an obstructed vent is opened improperly, 5 L of the solution is assumed to be ejected. The material drops from a 3 m height onto the floor outside a glovebox. This sequence of events results in a conservative estimate since the top of the tallest container assembly is 2 m, and the improper venting of the container would result in mostly gaseous products being released. Unlike a pressurized leak from a line under pump pressure, the pressure in the container would quickly decrease with time resulting in less material ejection than is assumed in the analysis. The consequences of this event bound all other potential radioisotope bearing liquid pressurization events.

As a result of a spill from a height of 3 m, it was estimated that 0.002% by weight of the plutonium nitrate solution as droplets will become airborne using NRC approved methodology presented in NUREG/CR-2139. This results in 0.16 g of Pu becoming airborne. Upon detection of the spill, the spill area is evacuated and confined to people with breathing apparatus for cleanup, limiting the exposure to facility personnel. Any airborne particulates are discharged into the E-3 or E-4 system from Zone 3 or 4 with filtration through at least a

single stage roughing filter and a single stage of HEPA filtration. The HEPA filters have an efficiency of 99.95% resulting in less than 8.0×10^{-5} g of Pu reaching the PFP stack.

Evaluation Findings. The possibilities for overpressurizing liquid containers have been reviewed. Several initiating events can lead to overpressurized containers resulting in the release of hazardous material. WHC found that the bounding overpressurization of a liquid container accident in regard to radiological consequences was to eject material from a 10-L container of plutonium nitrate solution. The scope of the review has included investigations of initial conditions, assumptions, and results of the overpressurization. The method used to determine the extent of airborne dispersal has also been examined and found acceptable. The worst case Pu release to the room is 0.16 g. Using one stage of HEPA filtration, the worst case Pu release is 8.0×10^{-5} g resulting in 2.8×10^{-5} rem EDE onsite and 1.0×10^{-5} rem EDE offsite. These results are well below the DOE requirements and the risk guidelines presented in section 9.4 for an annual event, and are therefore acceptable.

The HEPA efficiency is maintained by OSR 3.2.1, *Filtered Exhaust From Zones 3 and 4*, which specifies that the minimum acceptable efficiency for filtration of radioactive particles from the ventilation exhaust stream is 99.95%, and the minimum number of HEPA filter stages in the ventilation exhaust stream. The basis for acceptance in the review is that WHC's choice of maximum overpressurization of a liquid container accident has been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.3 Radioisotope-Bearing Powder Spills

Discussion. This event is forwarded from PA element E-1 (see PFP FSAR Section 9.0.2.4). Powder spills and subsequent airborne contamination have occurred in the past due to improper handling, inadequate training, and less than adequate design of containers. Normally, PuO_2 powder is contained in metal slip-lid cans capable of holding 1 kg of material. The slip-lid is taped to the can and the can is then removed from the glovebox in a heat sealed 8 mil-thick bag. The can and bag are then mechanically sealed into two successively larger food-pack cans. Smaller quantities of materials are also stored, but the 1 kg of material was found to be the largest quantity of Pu powder handled outside of confinement.

All radioisotope-bearing powders handled in the PFP are confined at all times within confinement systems such that a single failure will not result in release to the environment. Contaminated powders are handled within enclosed gloveboxes or hoods that have preferential airflow away from operational personnel. These powders are not brought out from confinement until they have been enclosed in several overpacks and in an outer can.

For the purposes of this analysis, WHC assumed that the entire contents of a 1-kg slip-lid can containing PuO_2 spilled from a height of 3 m onto the floor outside a glovebox. The 1-kg mass and the slip-lid can are conservative assumptions in that the 1-kg masses are sealed in

three successively larger containers, and plutonium powder is normally handled in 400 g amounts. It is also assumed that all other systems are operating normally prior to the spill. The consequences of this accident bounds all other possible radioisotope-bearing powder spills in the PFP.

As a result of a free fall spill from a height of 3 m, it is estimated that 0.12 weight percent of the PuO_2 will become airborne (using an NRC approved methodology presented in NUREG/CR-2139) resulting in 1.2 g of material immediately becoming airborne. Upon detection of the powder spill, the spill area is evacuated and confined to people with breathing apparatus for cleanup, limiting the exposure to facility personnel. Any airborne particulates are discharged into the E-3 or E-4 system from Zone 3 or 4 with filtration through at least one roughing filter and one stage of HEPA filtration. The HEPA filters have an efficiency of 99.95% resulting in less than 6×10^{-4} g of Pu reaching the PFP stack.

Evaluation Findings. The possibilities for spilling powder bearing containers have been reviewed. Several initiating events can lead to a powder spill resulting in the release of hazardous material. Westinghouse Hanford Company found that the bounding event in regard to radiological consequences was the spill of a 1-kg slip-lid can of PuO_2 . The scope of the review has included investigations of initial conditions, assumptions, and results of the spill.

Several conservative assumptions have been made in performing this analysis. In order for a spill to occur, the bag on the sealed slip-lid can would have to be punctured, and the slip-lid can would have to open and spill its contents. Any sharp object could easily puncture the bag, but the accidental opening of a slip-lid can has been demonstrated to be difficult. Drop tests from a height of 4.3 m on 25 slip-lid cans containing a variety of materials resulted in no failures. These tests were conservatively performed in that the tape seal normally placed over the can/lid joint was purposely left off. Therefore, at least two independent failures, puncture of over-bag and failure of slip-lid, are required for a powder spill.

The method used to determine the extent of airborne dispersal has been examined and found acceptable. The worst case Pu release into the room is 1.2 g. Using HEPA filtration, the worst case Pu release is 6.0×10^{-4} g resulting in doses of 1.8×10^{-4} rem EDE onsite and 6.2×10^{-5} rem EDE offsite. These results are well below the DOE requirements and risk guidelines presented in section 9.4 for an annual event, and are therefore acceptable.

The HEPA efficiency is maintained by OSR 3.2.1, *Filtered Exhaust From Zones 3 and 4*, which specifies that the minimum acceptable efficiency for filtration of radioactive particles from the ventilation exhaust stream is 99.95%, and the minimum number of HEPA filter stages in the ventilation exhaust stream. The basis for acceptance in the review is that WHC's choice of a maximum powder spill accident has been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.4 Radioisotope-Bearing Pressurized Powder Container

Discussion. The handling of containers of radioisotope-bearing powders which become pressurized both within and outside of process enclosures was reviewed for the potential to cause abnormal releases of contaminated material. Events that may lead to the pressurized discharge of radioisotope-bearing oxide powders and plutonium bearing recoverable materials are identified in PA element D-19 (see PFP FSAR Section 9.0.2.4). Radioisotope-bearing materials stored in closed containers can pressurize those containers due to the effects of radiolysis, chemical reactions, or internal thermal energy. Forms of plutonium bearing material include plutonium metal, dry powders, radioisotope-bearing recoverable material items, and plutonium nitrate solutions. The probability of container failure and the magnitude of any resulting contamination spread is increased by container pressurization. Personnel exposures may be incurred if pressurized containers rupture during handling or if rupture occurs outside confinement.

Radioisotope-bearing powders are stored in metal slip-lid cans capable of holding a mass of 1 kg. These containers are stored in successively larger cans which provide the assurance of a can-within-a-can protection against both inward and outward leakage. Plutonium metal ingots are also stored with this same methodology. Radioisotope-bearing recoverable material is first sampled and tested against loss-on-ignition (LOI) criteria. The material is found to be unstable if the amount of material loss to gas generation exceeds 1 wt%. Unstable recoverable material is stored in specified gloveboxes in the 234-5Z Building. Stable material is placed in either polyethylene jars or slip-lid cans, and then placed in gloveboxes. The quantity of recoverable material in each container, and position in the gloveboxes, is administratively controlled to provide criticality safety. All containers are inspected monthly for pressurization with containers showing the effects of pressurization being repackaged.

For the purposes of this analysis WHC assumed a failure to evaluate and control moisture when a plutonium metal ingot was packaged. This results in PuO_2 formation which leads to container pressurization and failure. Since the plutonium metal oxidizes slowly, the principle release mechanism was assumed to be the slow oxide growth resulting in the slow release of powder which falls to the floor below.

The effects of a can rupture were modeled from data generated from pressurized releases, free fall powder spill experiments, and resuspension data. The total amount of suspended material will only be a few grams and will be bounded by the consequences of the powder spill analysis discussed in Section 9.1.3. Upon detection of the material release, the spill area is immediately evacuated and confined to people with breathing apparatus for cleanup. Also, personnel involved with plant or process operations that may result in airborne radioactivity are required by plant operating procedures to wear respiratory protection limiting exposure to facility personnel. Any airborne particulate's are vented through the E-3 filter resulting in the activity at the stack being below detection levels.

Evaluation Findings. The possibilities for the pressurization and failure of powder-bearing containers have been reviewed. A number of initiating events can lead to the pressurization of plutonium bearing containers resulting in the release of radioactive material. This review determined that the bounding event in regard to radiological consequences was the pressurization of plutonium metal ingot containers due to metal oxidation to PuO_2 powder. The scope of the review has included investigations of initial conditions, assumptions, and results of the container failure.

The method used to determine the extent of airborne dispersal has been examined and found acceptable. This event is bounded by the powder spill event in Section 9.1.3 so the worst case Pu release is 6.0×10^{-4} g resulting in 1.8×10^{-4} rem EDE onsite and 6.2×10^{-5} rem EDE offsite. These results are well below the DOE requirements and risk guidelines presented in section 9.4 for an annual event, and are therefore acceptable.

The HEPA efficiency is maintained by OSR 3.2.1, *Filtered Exhaust From Zones 3 and 4*, which specifies that the minimum acceptable efficiency for filtration of radioactive particles from the ventilation exhaust stream is 99.95%, and the minimum number of HEPA filter stages in the ventilation exhaust stream. The basis for acceptance in the review is that WHC's choice of maximum container pressurization accident has been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.5 Stack Release Resulting From Filter Changes

Discussion. The potential for a stack release resulting from errors made during change out of secondary HEPA filters is forwarded from PA element D-4 (see PFP FSAR Section 9.0.2.4). The secondary HEPA filters are the last barrier between contamination and the stack. Other events associated with the HEPAs, such as a filter fire, were determined to be incredible due to the small radionuclide loading in the filters. WHC postulated for this analysis that during a routine filter change in Room 309 or 310 in the 234-5Z Building, a HEPA filter is dropped 8 ft onto a concrete floor. This results in the release of contaminated dust into the filter room. Because the louvered outlet dampers do not close completely, the plutonium laden dust would be drawn to the exhaust plenum and out the stack.

Westinghouse Hanford Company assumed in the filter change analysis that the louvered dampers have been closed to isolate the room from the stack plenum, but no plastic cover is placed over them as required by plant operating procedures to prevent leakage flow up the stack. The filters are assumed to contain ten times as much plutonium as was measured in the filters after ten years of service. Therefore, it is assumed that 0.9 g of plutonium is contained in the filter. Since both assumptions are conservative with respect to plant procedures and measured filter amounts, these assumptions are considered acceptable for this analysis. After the filter is dropped, the filter change continues until the 291-Z Building stack continuous air monitor (CAM) alarms, and the work crew is notified, approximately 30 minutes.

The consequences of the dropped filter are limited to the amount of plutonium laden dust released when the filter is dropped. Several conservative studies have determined that 1% of the inventory of a dropped HEPA filter would be released as respirable particles. Thus, 9×10^{-4} g of respirable plutonium would be released up the stack. This amount of plutonium is well above the amount needed to trip the stack CAM alarm, but a factor of 10 below the release limit for an event with an annual probability of one. Facility personnel who are changing filters are required by plant operating procedures to wear respiratory protection limiting their exposure.

The analysis of a dropped filter in Room 309 or 310 within the 234-5Z Building was determined to be the bounding dropped filter event. Westinghouse Hanford Company also postulated that a dropped filter could occur in Room 26 of the 236-Z Building. In performing the analysis, it was shown that the plutonium inventory is less in these filters than in the filters in the 234-5Z Building. Therefore, since the release from the 236-Z-1 stack would be considerably less than 0.009 g respirable plutonium, the consequences of this event are bound by the dropped filter in Room 309 or 310 event.

Evaluation Findings. The possibilities for the release of plutonium laden dust from dropping a HEPA filter have been reviewed. Two initiating events could result in the release of radioactive material. WHC found that the bounding event in regard to radiological consequences was a dropped filter in Room 309 or 310 within the 234-5Z Building. The scope of this review has included investigations of initial conditions, assumptions, and results of the container failure.

The method used to determine the extent of airborne dispersal has been examined and found acceptable. The worst case Pu release from the 291-Z-1 stack is 9.0×10^{-3} g resulting in 3.2×10^{-3} rem EDE onsite and 1.1×10^{-3} rem EDE offsite. These results are well below DOE requirements and the risk guidelines presented in section 9.4 for an annual event, and are therefore acceptable. The basis for acceptance in the review is that WHC's choice of a maximum dropped filter event has been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.6 Overpressure of Remote C Line Pressure Vessel

Operation of the RMC line is not required for completing PFP's currently planned activities. If in the future it is determined that operation of the RMC line is necessary, then WHC shall submit a safety analysis for RMC line operations to DOE for review and approval.

9.1.7A Excessive Building Pressurization -- 234-5Z

Discussion. The ventilation system air establishes four levels of pressure for rooms within the building to maintain contamination control. The rooms and areas which contain radioactive materials are maintained at a pressure below the surrounding atmospheric pressure.

Office areas, free of radioactive materials, are maintained at a pressure above surrounding atmospheric pressure. Rigid barriers separate this zone from the other zones. An airlock is used to pass between areas. This system is controlled by five dampers in the various airstreams. This analysis determined the consequences of pressurizing the rooms in the 234-5Z Building which contain radioactive sources. If the pressure level in any of these rooms is greater than the surrounding atmospheric pressure, any loose radioactive materials will tend to be forced to the outside environment upon breach of outer confinement.

For the purpose of this analysis, WHC assumed that the building would become partially pressurized by the opening of one or more of the supply dampers attached to the supply plenum controlling the discharge of air to the various zones. The malfunction of the dampers would be due to inadvertent opening (or closing) of a valve on the pneumatic control lines. Numerous small control valves are associated with the control system for the dampers. The probability of an inadvertent opening (or closing) of a valve on the pneumatic control lines is given a probability of occurrence of one.

The inadvertent opening of a damper control valve would result in the loss of control pressure and the damper would fail open. The airflow to the system controlled by the damper would increase, increasing pressure in affected rooms. Analyses were performed for failure of damper C, damper B, dampers C and B, damper D, and damper E. Failure of damper A was not discussed because zone 1, supplied by damper A, is maintained at positive pressure with respect to atmospheric pressure and all other zones. A failure of damper A would only increase the positive pressure a small amount and would not affect contamination control.

Simplified calculations were performed for all of the failed damper scenarios (C, B, C&B, D, and E). These calculations showed that opening of the affected damper(s) would briefly change the ventilation pressure for those rooms throughout zone 3 that receive air solely from the failed damper(s) airstream. Pressure would increase from the normally negative pressure to positive pressure. The rooms serviced by the other dampers would experience a drop in pressure making them more negative. The positive pressure in rooms containing contaminated material can lead to the uncontrolled release of radioactive material to clean rooms or the environment. However, as was shown in the spill events discussed in Section 9.1.1 through 9.1.4 of this SER, there is not enough free material that could be released to exceed radioactive material release acceptance limits for on-site and off-site exposures. The consequences of the spill events bound any contamination spread from a failed damper(s). The increased pressurization would not affect the structural capacity of the wall material and therefore would not cause the wall confinement to fail which would result in direct release to the environment.

Upon notification of the ventilation upset condition from alarms or plant operating personnel, the power operations personnel would verify the upset conditions and then follow plant operating procedures for the identified upset condition and trip to the steam driven exhaust fans (reference Appendix A). The exhaust fans return the building to negative pressure and maintain radiological contamination control. Upon tripping to the steam driven exhaust fans,

personnel in the 234-5Z Building are evacuated, thus limiting any potential for exposure. Any personnel remaining in the building for shutdown or cleanup are required by facility procedures to wear respiratory masks.

Evaluation Findings. The possibilities for excessive pressurization of the 234-5Z Building have been reviewed. Several initiating events can lead to the pressurization of the building resulting in the potential release of radioactive material. These events include failure of damper C, damper B, dampers C&B, damper D, and damper E. This review determined that the consequences of all events are bounded by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The scope of the review has included investigations of initial conditions, assumptions, and results of the inadvertent closure of a damper(s) control valve.

The calculational method and assumptions used to determine the amount of room pressurization resulting from damper(s) failure was examined and found acceptable. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The method used to determine the extent of airborne dispersal has been examined and found acceptable. The onsite and offsite consequences of the spill events bound any contamination spread from a failed damper(s).

The spread of contamination to other areas of the facility is limited by confinement design features. The HEPA efficiency is maintained by OSR 3.2.1, *Filtered Exhaust From Zones 3 and 4*, which specifies that the minimum acceptable efficiency for filtration of radioactive particles from the ventilation exhaust stream is 99.95%, and the minimum number of HEPA filter stages in the ventilation exhaust stream. The basis for acceptance in the review is that WHC's choices for 234-Z Building pressurization events have been confirmed, that the methods and assumptions used in the analyses are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.7B Excessive Pressurization of 236-Z Building

Discussion. This analysis examined the mechanisms for pressurizing the areas in the 236-Z Building containing radioactive materials. Ventilation air for the 236-Z Building is taken from the ventilation system for the 234-5Z Building. Damper F is attached to the supply plenum within the 234-5Z Building and controls ventilation air flow. A cascading system of air usage is used for contamination control. At each successive point in the building, air pressure is less than at the previous location (greater negative pressure). All rooms are maintained at a negative pressure with respect to atmospheric pressure for contamination control.

The entire 236-Z Building, including production areas, can be pressurized by opening damper F. For the purpose of this analysis, it was assumed that damper F failed in the open position due to human error during routine maintenance. Failure of this damper would result in airflow to the 236-Z Building, increasing the building pressure to positive with respect to the

atmospheric pressure. The drop in pressure in the supply plenum is compensated for by the modulation system on the supply fans.

A simplified calculation was performed for the failed damper scenario. This calculation showed that opening the affected damper would change the ventilation pressure from -0.15/-0.3 in. w.g. to +0.35/+0.20 in. w.g. Pressure would increase from the normally negative pressure to positive pressure. The positive pressure in rooms containing contaminated material can lead to the uncontrolled release of radioactive material to clean rooms or the environment. However, as was shown in the spill events discussed in Section 9.1.1 through 9.1.4 of this SER, there is insufficient free material that could be released to exceed radioactive material release acceptance limits for on-site and off-site exposures. The consequences of the spill events bound any contamination spread from the failed damper. The increased pressurization would not affect the structural capacity of the wall material and therefore would not cause the wall confinement to fail, which would result in direct release to the environment.

The only area where radiological contamination may be a potential problem is in the airlock adjoining the canyon. Currently, the airlock is occasionally found to have spots of contamination. Operating personnel believe that this contamination seeps from around the canyon door into the airlock. When pressurization occurs due to failure of damper F, there might be a slight increase in the amount of seepage into the airlock. However, this presents only a facility cleanup problem because the airlock is sealed against the outside of the building (the increase in pressure does not approach that necessary to affect wall stability). A CAM alarms when the airlock becomes contaminated, alerting operations staff to the pressurization upset condition.

Upon notification of the ventilation upset condition from alarms or plant operating personnel, the power operations personnel would verify the upset conditions and then follow plant operating procedures for the identified upset condition and trip to the steam driven exhaust fans. The exhaust fans return the building to negative pressure and maintain radiological contamination control. Upon tripping to the steam driven exhaust fans, personnel in the 236-Z Building are evacuated, limiting any potential for exposure. Any personnel remaining in the building for shutdown or cleanup are required by facility procedures to wear respiratory masks.

Evaluation Findings. The possibility for excessive pressurization of the 236-Z Building has been reviewed. The initiating event that leads to pressurizing the building is the failure of damper F resulting in the potential release of radioactive material. This review determined that the radiological consequences of this event are bounded by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The scope of the review has included investigations of initial conditions, assumptions, and results of the inadvertent opening of damper F.

The calculational method and assumptions used to determine the amount of room pressurization resulting from damper failure was examined and found acceptable. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The method used to determine the extent of airborne dispersal has been examined and found acceptable. The onsite and offsite consequences of the spill events bound any contamination spread from a failed damper(s).

The spread of contamination to other areas of the facility is limited by confinement design features. The basis for acceptance in the review is that WHC's choice for the 236-Z Building pressurization event has been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

9.1.7C Excessive Building Pressurization - 2736-ZB Building

Discussion. This analysis examined the mechanisms for pressurizing areas in the 2736-ZB Building that contain radioactive source material. The ventilation for the 2736-ZB Building is self-contained and independent of the other ventilation systems servicing the PFP. The instrumentation is considerably more extensive and sophisticated. The system uses computers to monitor and control system operating parameters.

The ventilation system is actually composed of two independent systems which share the same exhaust stack. System 1 pumps air into the building and into an air duct, which then separates into two streams. The total system 1 flow recombines and is filtered before release out the stack. System 2 also pumps air into the building and an air duct. However, the air sweeps through only the non-destructive analysis (NDA) room and then into a separate filter system before release out the stack.

Pressure is controlled through controllable dampers on both the supply and exhaust side. The building pressure is maintained negative with respect to atmospheric pressure to maintain contamination control. Three scenarios were proposed to accidentally pressurize the building. WHC assumed for the purposes of this analysis that 1) the supply control damper has failed open or has been driven open, 2) the exhaust fan has failed, or 3) the exhaust control damper has failed closed or driven closed.

The supply fan dampers are designed to fail closed on loss of pneumatic pressure and loss of power or instrument air. Failure of the supply fan dampers is not a credible accident initiator, but the configuration of the exhaust fans could lead to a potential event. The repackaging room can be pressurized due to failure of one exhaust fan when the other fan is out of service for routine maintenance. The bounding pressurization event, however, is when the exhaust control damper fails closed when the exhaust fans are in manual control. Loss of control air would allow the dampers to fail closed, but the fans would not respond to the generated shutdown signal because they are in manual control. This would pressurize the repackaging room and most of the east side of the building.

The possibility of radioactive materials being released to the environment will be remote for this event. The discussions in Sections 9.1.1 through 9.1.4 bound the radiological consequences for this event. These sections show that there is not a potential for significant radiological consequences to either the public or facility staff. The upset condition would be immediately signaled by alarms on the annunciator board in the control room. The power operators would then follow plant operating procedures to maintain contamination control.

Evaluation Findings. The possibility for excessive pressurization of the 2736-ZB Building has been reviewed. The bounding initiating event that leads to pressurizing the building is for the exhaust control damper to fail when the exhaust fans are in manual control, resulting in the potential release of radioactive material. This review determined that the radiological consequences of this event are bounded by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The scope of the review has included investigations of initial conditions, assumptions, and results of the event.

The calculational method and assumptions used to determine the amount of room pressurization resulting from damper failure was examined and found acceptable. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The methods used to determine the extent of airborne dispersal and the consequences of that dispersal have been previously examined and found acceptable in Sections 9.1.1 through 9.1.4 of this SER.

9.1.7D Excessive Building Pressurization - 2736-Z Building

Discussion. The postulated events which lead to building pressurization are the same as the 2736-ZB analysis discussed in Section 9.1.7C of this SER. The possibility of the release of radioactive materials to the environment will be remote for this event. The building structure is able to withstand the pressure increase, and no loose contaminated material is stored in the facility. Therefore, the discussions in Sections 9.1.1 through 9.1.4 bound the potential radiological consequences for this event. These sections show that a potential for significant radiological consequences does not exist to either the public or facility staff. The upset condition would be immediately signaled by alarms on the annunciator board in the control room. The power operators would then follow plant operating procedures to maintain contamination control.

Evaluation Findings. The possibility for excessive pressurization of the 2736-Z Building has been reviewed. The bounding initiating event that leads to the pressurization of the building is the failure of one exhaust fan when the other is out of service, and the control dampers fail closed, resulting in building pressurization. This review determined that the radiological consequences of this event are bounded by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The scope of the review has included investigations of initial conditions, assumptions, and results of the event.

The calculational method and assumptions used to determine the amount of room pressurization and resulting pressurization of the 2736-Z Building were examined and found acceptable. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The methods used to determine the extent of airborne dispersal and the consequences of that dispersal have been previously examined and found acceptable in Sections 9.1.1 through 9.1.4 of this SER.

9.1.7E Excessive Building Pressurization - 241-Z Building

Discussion. Unlike the previous buildings discussed above, the 241-Z Building ventilation does not use a supply system. Ventilation air results from in-leakage into the building as a result of atmospheric pressure. One exhaust ventilation system with two branches serves both the 241-Z tanks and cells. Exhaust air is heated to protect the HEPA filters from moisture buildup, and routed through two stages of HEPA filtration, in series, before discharge to the atmosphere via the 296-Z-3 stack. Ventilation upsets that could be caused by a ventilation abnormality, a break in a the service line, or a chemical reaction in a storage tank were reviewed.

A set of conditions were not identified by WHC which could result in a ventilation upset condition. Westinghouse Hanford Company safety analysts reviewed the possibility of two service lines being pressurized, one line with steam, the other with air. Both lines could be pressurized by valving from outside the vault confines. A break would have to occur when the lines were energized. Any break would be noticed quickly and rectified because of tank monitoring for (NO)_x gas which is generated in the D-5 tank. A break at any other time would not discharge pressurized gas since the lines are normally de-energized. Under worst case conditions, it is expected that a gas flow of 36 ft³/min for 6.6 minutes could be generated. The exhaust system for the 241-Z Building is capable of 1,000 ft³/min. Therefore, there is no potential for pressurization of the building.

Evaluation Findings. The possibility for excessive pressurization of the 241-Z Building has been reviewed. The scope of the review has included investigations of initial conditions, assumptions, and results of the event. It was determined from the review that a credible event does not exist that can result in the pressurization of the 241-Z Building that would lead to a loss of contamination control.

9.1.7F Building Depressurization - All PFP Operating Buildings

Discussion. The possibility for excessive depressurization of all the PFP operating buildings has been reviewed. WHC determined that postulated depressurization events in the 236-Z, 2736-ZB, 2736-Z, and 241-Z are either not credible or do not affect contamination control and would not result in the release of any contaminated materials. However, a possibility does exist that a depressurization event could spread loose contaminated dust in the 234-5Z Building.

Two possible events were investigated as possible mechanisms for depressurization in the 234-5Z Building: valve adjustment errors and icing over of the humidifier pad on the supply unit air washers. Inadvertent adjustments of the valves on the pneumatic control lines leading to damper B and C could cause localized loss of control air, resulting in the dampers failing open. The ventilation control system would attempt to correct the resulting overpressure (as described in Section 9.1.7A), causing depressurization only in the rooms serviced by the working dampers. The second event results in depressurization of the supply plenum which causes a depressurization of the building.

Depressurization of certain rooms by the failure of dampers B or C could expose certain of these rooms to potential airborne radioactivity from other rooms. If the adjoining doors are not securely latched, a rush of air would move from high to low pressured rooms. However, the potential radiological consequences of this spread of contaminated material are bounded by the spill events (Sections 9.1.1 through 9.1.4) discussed early in this Chapter.

A depressurization of the building through freezing of the supply unit air washer humidifier pads is a slower transient. No severe pressure differentials that drive the spread of contaminated material (as in the damper failure case) could build up between rooms. Also, no potential exists for gloveboxes to spread their contents because the design allows the glovebox pressures to remain negative with respect to the room pressure, whether the room is overpressurized or depressurized.

The discussions in Sections 9.1.1 through 9.1.4 bound the potential radiological consequences for this event. These sections show that no potential for significant radiological consequences exists to either the public or facility staff. The upset condition would be immediately signaled by alarms on the annunciator board in the control room. The power operators would then follow plant operating procedures to maintain contamination control.

Evaluation Findings. The possibility for depressurization of the PFP operating buildings has been reviewed. The bounding depressurization event was determined to be in the 234-5Z Building. This review determined that postulated depressurization events in the 236-Z, 2736-ZB, 2736-Z, and 241-Z are either not credible or do not affect contamination control and would not result in the release of any contaminated materials. It was found that any radiological consequences of this event are bounded by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The scope of the review has included investigations of initial conditions, assumptions, and results of the event.

The assumptions used to determine room depressurization of the 234-5Z Building were examined and found acceptable. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The methods used to determine the extent of airborne dispersal and the consequences of that dispersal have been previously examined and found acceptable in Sections 9.1.1 through 9.1.4 of this SER.

9.1.8 Loss of Ventilation

Discussion. A loss of ventilation event occurs when both supply and exhaust fans lose power simultaneously. Air in the facility will become virtually stagnant except for a slight reverse air flow that can be generated due to convection or for pressure drops resulting from weather changes. All facilities have similar responses upon loss of ventilation, but the buildings serviced by the 291-Z-1 stack are bounding. In general, during a loss of ventilation event, the only airflow in the building would be from natural circulation. In all but very hot weather, convection can provide a significant draft, as can the suction that results from wind blowing across the lip of the 200-ft high stack, thus maintaining the normal airflow direction. Two situations, however, can potentially result in reversal of airflow into the building. Reverse flow down the 291-Z-1 stack could occur as a result of 1) summer heat induced convection or 2) winter weather induced pressure drop from a passing storm front.

For the purposes of this analysis, it is assumed that a complete loss of ventilation occurs due to a simultaneous loss of electricity and steam. Electrical loss could be local to the building or site-wide. Steam loss occurs due to a broken steam pipe or failure at the online steam plant. A complete loss of ventilation would require three simultaneous, unlikely events, i.e., loss of two electrical circuits and steam. Upon loss of power, the supply fan damper opens, supply fans stop, dampers B through E open to their failed position, and dampers A and F fail closed, isolating Zones 1 and 3B from the rest of the system. This sequence of events leaves an open path from the supply louvers through the Zone 3 rooms. Also, an air path is established through all room ventilated gloveboxes. The supply dampers for the dry air system gloveboxes close, preventing flow through the dry gloveboxes. The loss of ventilation is immediately detected by an alarm in room 321 and all the personnel in the Zone 3 areas are evacuated to Zone 1 except those needed for the orderly shutdown, and they don respiratory protection.

(1) Flow Reversal - Hot Day

For this case, it is assumed that the outside air temperature is 110°F, the inside air temperature is 80°F in buildings 234-5Z and 236-Z, and it is windless. Wind would have the effect of drawing air up the stack due to the low pressure produced at the top of the stack from the Bernoulli effect. To obtain backflow down the stack, the air in the building must backflow through the ducting into the supply plenum and up into the supply fans. From the fans, the air would backflow through both the wet and dry filters to exit from the fan inlet grilles. The maximum backflow rate is about 15,000 ft³/min, as compared to a normal forward flow of about 225,000 ft³/min.

(2) Flow Reversal - Cold Day

For this case, it is assumed that a sudden change in atmospheric pressure occurs while the building ventilation is out, and the building has been without heat for a considerable time. Only after the reservoir of heat stored in the building structure and equipment cooled would

thermal convection cease to force air up the stack. The average atmospheric pressure at the Hanford site is 29.21 in. Hg. From weather data collected over the past 40 years, the maximum hourly pressure drop was found to be 0.160 in. Hg, which lowers the atmospheric pressure to 29.05 in. Hg. The maximum backflow rate is assumed to be 60,000 ft³/min. (A more accurate calculation was performed by the review team which determined the maximum backflow rate to be 18,600 ft³/min, refer to Appendix A).

The radiological consequences of both flow reversal scenarios are not significant. No significant source term or internal to building contamination follows either the convection or atmospheric pressure drop induced flow. This is due to the presence of HEPA filtration, building confinement integrity, and the lack of a meaningful driving force to push plutonium out of the primary confinements. A possibility exists for removing material normally downstream of open faced hoods. The consequences of this possibility are bounded by the spill events discussed in Section 9.1.1 through 9.1.4 of this SER.

A possibility of hydrogen gas accumulation exists in tanks containing plutonium solutions as a result of radiolysis. The vessel vent systems direct this and other gases to the gloveboxes and then to the E-4 vent system. Loss of ventilation would increase the hydrogen gas levels in the gloveboxes. Upon review of this event, it was determined that for the worst case scenario, it would take from 3 to 5 months to accumulate enough hydrogen gas to result in a structure threatening explosion; therefore, there is ample time to either restore ventilation flow or provide for other corrective action to ensure that any potential explosion is averted.

Loss of ventilation in the 2736 complex would also affect cooling to the containers of plutonium stored in the vault. Calculations in section 9.2.4A.6.3.6.D of the FSAR show temperatures of about 280°F and a pressure of 20 psi after 48 hours. The storage cans would not burst under these conditions. Loss of ventilation in the rest of the 2736 Buildings would have similar results as in the 234-5Z Building.

Evaluation Findings. The possibility for loss of ventilation in the 234-5Z and the 236-Z Buildings has been reviewed. The two initiating events that can lead to the reversal of airflow in the building are convection on a hot day and drop in atmospheric pressure due to weather conditions. It was found that the radiological consequences of these events are bound by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The potential for hydrogen gas accumulation and deflagration, and movement of entrained Pu powder were also reviewed. The scope of these reviews included investigations of initial conditions, assumptions, and results of the loss of ventilation and flow reversal.

The calculational method and assumptions used to determine the amount of airflow reversal resulting from a hot day and winter weather induced atmospheric pressure drop were reviewed. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The methods used to determine the amount of hydrogen gas generation during the loss of ventilation event was also reviewed and found appropriate for the level of potential consequence.

A margin of protection pertaining to these events is contained within the ventilation system itself. The ducting is of varied geometric shapes and sizes routed through the building with many bends, and several banks of HEPA filters are in place. These features combine to offer a large resistance to any backflow of air that may occur. Since the airflow takes the path of least resistance, any backflow generated will be directed to the E-3 system, which supplies the rooms. The E-4 system includes several stages of HEPA filterboxes, as compared to one stage in the E-3 system.

Loss of ventilation in the 2736 complex would also affect cooling to the containers of plutonium stored in the vault. Calculations in section 9.2.4A.6.3.6.D of the FSAR show temperatures of about 280°F and a pressure of 20 psi after 48 hours. An independent review of the heat transfer analysis of the container heat-up was performed. It was determined that the WHC analysis was conservative. Therefore, WHC has demonstrated that loss of ventilation to the vault does not result in a significant safety issue.

A potential problem with entrained plutonium powder downstream of the final HEPA filters was identified in the FSAR. There is 55,000 ft³ of ventilation system duct work downstream of the final stage of HEPA filters that leads to the 291-Z-1 exhaust stack. This duct work has been contaminated with material conservatively estimated to contain up to 20 g of plutonium powder from operations over the past 40 years. It was originally postulated that this powder may be loose and has collected in certain areas of the ventilation system, held in place by only air pressure. However, recent studies performed by both WHC (FitzPatrick 1989 and PFP Analytical Laboratory 1988) and PNL (Mahoney 1993) have demonstrated the following:

- (1) Air flow simulations in the ventilation system have demonstrated that from dead air to full ventilation flow takes approximately 30 seconds, with no significant difference between peak transient airflow velocity and steady-state airflow velocity. Therefore, the transient air velocity is not large enough to produce an adequate driving force to dislodge any plutonium particles adhering to the ducts.
- (2) Normal PFP operation requires that the operating exhaust fans be rotated between seven electric exhaust fans. The two steam driven fans are also routinely tested. The ventilation system has also been driven by only one steam driven exhaust fan (normally four electric driven fans are used) due to an operational event. These operational events, rotations, and tests regularly change the air flow patterns in the ventilation system. To date, these changes have not resulted in measurable effluent releases, as detected by the 291-Z-1 CAM. This demonstrates that there are not a substantial quantity of loose particles that can be released from the stack.

Based on the above evidence, WHC has demonstrated that plutonium particles located past the final HEPA filters have adhered to the duct walls and are not affected by loss of ventilation conditions and, therefore, do not result in a significant safety issue.

The method used to determine the extent of airborne dispersal has been examined and found acceptable. Westinghouse Hanford Company has also met the requirements that co-located onsite exposure be limited to 0.5 rem EDE and offsite exposures be limited to 0.01 rem EDE for an abnormal event with an assumed frequency of once per year. The basis for acceptance in the review is that the WHC choice for the 234-5Z and 236-Z Building pressurization events and the vault heat-up event have been confirmed, that the methods and assumptions used in the analysis are conservative, and that specified acceptable radiological dose limits will not be exceeded.

The accident analyses for the PFP have not identified any scenarios where continued operation of the ventilation system is essential to prevent an unacceptable airborne release of plutonium. This section and Sections 9.1.7 and 9.2.2 of the FSAR provide a comprehensive overview of the ventilation system as it relates to safety issues. For the analysis presented in these sections, the building confinement provides the ultimate barrier to the release of plutonium. Therefore, a static air posture is an acceptable facility situation.

9.1.9 Glovebox Glove Breach

Discussion. Penetration of a glovebox glove creates an exposure path for releasing plutonium bearing material to areas occupied by facility personnel. Factors that can lead to glove failure include hot surfaces, sharp objects, and deteriorated glove material. A breached glove would be a violation of primary confinement, but would not necessarily cause plutonium bearing material to be released. The object of this section is to review the conditions and scenarios that could and would result both in violating primary confinement and releasing plutonium bearing material into an occupied space.

Westinghouse Hanford Company assumed in analyzing this event that sources such as hot surfaces, a sharp object, or deteriorated glove material lead to glove failure. A glove breach violates the primary confinement barrier between operational personnel and a contaminated environment. For a release to occur as a result of the glove breach, the energy source which penetrates the glove must also transport the contaminates out while the normal glovebox negative pressure is missing or reversed. Several explosion scenarios can result in glove failure by any object of significant mass and velocity penetrating the glovebox, e.g., a missile generated by an explosion or a rapid release of gas pressure. These types of events are discussed in Section 9.2, *Accidents*.

A breach in a glove will usually be detected by the operator or identified during routine radiation surveys. The event that has the most significant consequence is a puncture wound where the penetrating object carries radioactive materials into the tissues. This and other events can lead to a spread of contaminated material into occupied working areas. The total amount of Pu possible for release is bounded by the powder/liquid spill events discussed previously. Any airborne releases to onsite or offsite receptors is limited by building confinement barriers and the E-3 or E-4 ventilation system. The consequences of such an event are bounded by the analysis performed in Sections 9.1.1 through 9.1.4 of the FSAR.

All gloves known to have failed or to be leaking are replaced prior to performing glovebox operations per OSD 12.2.A. Also, plant operating procedures require that all glovebox gloves be replaced on a yearly basis as outlined in WHC-CM-5-8, procedure 1.33 *Glove Control System*. Glovebox pressures are also maintained such that airflows are toward more highly contaminated areas to be filtered (OSDs 12.2.B and 12.2.C). All glovebox negative pressures and exhaust filter pressure drops are verified to meet safety standards before entry into any glovebox for testing or maintenance. The following actions are also proceduralized to minimize the probability of internal deposition; 1) using leather gloves when handling plutonium bearing material, 2) training personnel to use special care when handling sharp objects or working near equipment with sharp edges, 3) inspecting containers for signs of internal pressure buildup, and 4) using filter masks when handling containers suspected of containing reactive material or having internal pressure.

Evaluation Findings. The possibility for the breach of a glovebox glove has been reviewed. Any radiological consequences of this event are bounded by the consequences of the spill events reviewed in Sections 9.1.1, 9.1.2, 9.1.3, and 9.1.4 of this SER. The scope of the review has included investigations of initial conditions, assumptions, and results of the events.

The assumptions used to determine a glovebox glove breach were examined and found acceptable. The simplified approach taken in the WHC analysis was determined to be appropriate for the level of potential radiological consequence. The methods used to determine the extent of airborne dispersal and the consequences of that dispersal have been previously examined and found acceptable in Sections 9.1.1 through 9.1.4 of this SER.

9.2 ACCIDENTS

Westinghouse Hanford Company selected accidents to be analyzed in Section 9.2 of the FSAR based on the resulting onsite/offsite consequences and the perceived notion of an accident (i.e., explosions). The following general accident categories were analyzed: explosions, fires, criticalities, seismic upsets, hazards associated with strong winds, aircraft accidents, and a carbon tetrachloride release. For each general category a number of specific accident scenarios were identified and analyzed.

- (1) Explosions - Two events were analyzed that could potentially result in an explosion. The first event is a decomposition of tributyl phosphate by nitric acid or heavy metal nitrates in the MT-5 dissolver pots. The second event is related to the potential hazards associated with propane gas bottles used at the facility.
- (2) Fires - The transients analyzed included a fire in the RMC line and a fire on the roof. The RMC line fire was determined to be the bounding fire scenario for the PFP facility.

- (3) Criticality - A generic criticality accident was performed to bound all possible criticality events in the PFP facility to define the criticality safety envelope. This generic approach is consistent with the methodology developed in NUREG-3.35.
- (4) Seismic - The seismic analyses performed for the PFP facility included building releases, and seismically induced criticalities, fires, explosions, and toxic releases. The seismic event is the most limiting accident scenario.
- (5) Strong Winds - The hazards associated with strong winds were reviewed for all of the building structures that compose the PFP facility. It was determined that outside nitric acid holding tanks, 2735-Z, and 55 gallon drums containing carbon tetrachloride located along the fence line were adversely affected by strong wind.
- (6) Aircraft Accident - The possibilities for aircraft accident events were evaluated. It was determined that the possibility for an aircraft accident involving the PFP was incredible based on methodology developed in NUREG-0800.
- (7) Carbon Tetrachloride Release - A carbon tetrachloride release was postulated to result from a truck crashing into 55-gallon drums stored within the fenced area of PFP. The release was shown to be the most limiting toxicological hazard for the onsite receptor.

9.2.1 Explosions

Four explosive hazards (red oil, hydrogen gas, flammable gas stored in bottles, and miscible organics) were identified in WHC's preliminary hazards assessment. However, as a result of changes in plant operational requirements only two scenarios were analyzed. The first scenario involves the decomposition of tributyl-phosphate (TBP) in the presence of heavy-metal nitrates (at elevated temperatures), and may result in the deflagration of combustible gases generated in glovebox MT-5. The second scenario identified a mishap involving propane gas bottles. Accident scenarios involving explosive hazards associated with plant operations involving the product concentrator, filtrate evaporator, laboratory waste concentrator, fluorinator, and slag and crucible system deentrainer have been removed from the PFP FSAR since the equipment is not currently operated at PFP. Westinghouse Hanford Company has committed to perform such safety analyses in the event that future situations warrant the use of this equipment, until such time the equipment will not be operated.

The term "red oil" is generally used to describe an organic material that can appear during concentration operations involving organic-TBP diluent and HNO_3 -Heavy Metal solutions. At elevated temperatures, heavy-metal nitrate-TBP complexes can exothermically decompose and generate flammable gases such as butene. If the rate of reaction is such that heat is generated faster than it can be removed then the reaction becomes self-sustaining. If the system is not vented the rapid production of gases may pressurize the system causing a catastrophic failure of the system and possible explosion. If the system is vented, the gases generated by the reaction are released to the surrounding area. If the concentration of flammable gases is

above the lower explosive limit the gases may be ignited and result in an explosion. The vented case is similar to the condition that exists in the dissolver pots in glovebox MT-5.

9.2.1A Product Concentrator

Currently there are no production plans or operational reasons to start up the product concentrator and, therefore, this accident scenario has been omitted from the PFP FSAR. Limiting Condition for Operation (LCO) 3.4.1.1 requires that the product concentrator remain shutdown. If the plant's operational mission changes such that it is necessary to operate the product concentrator, WHC shall perform a safety assessment and submit it to DOE for review.

9.2.1B Filtrate Evaporator

Currently there are no production plans or operational reasons to start up the filtrate evaporator and, therefore, this accident scenario has been omitted from the PFP FSAR. Limiting Condition for Operation (LCO) 3.4.2.1 requires that the filtrate evaporator remain shutdown. If the plant's operational mission changes such that it is necessary to operate the filtrate evaporator, WHC shall perform a safety assessment and submit it to DOE for review.

9.2.1C Deflagration in Glovebox MT-5

The original MT-5 safety analysis presented in the PFP FSAR (Draft 1991) was determined to be unacceptable, therefore, the MT-5 accident safety analysis became one of the nine open issues which required resolution. The initial safety analysis concluded that a seismically qualified MT-5 hot-plate power cut-off switch should be installed to ensure that the power to the MT-5 dissolvers would be secured in the event of an earthquake. The switch was never installed and, therefore, the safety analysis was invalid. As a result, WHC released a draft of WHC-SD-CP-SAR-012 Rev 2, *MT-5 Safety Assessment* (October 1993). This safety assessment included a more detailed and rigorous analysis of the chemical processes occurring in the MT-5 dissolver pots. The analysis incorporated new laboratory studies which provided experimentally derived reaction rates relating to the chemistry in the MT-5 dissolver pots. After several internal reviews by WHC's Safety Evaluation Advisory Council (SEAC), and informal DOE reviews, a final MT-5 safety assessment document was issued in June 1994. The DOE reviewers concluded that the safety assessment presented in WHC-SD-CP-SAR-012 Rev 2 was appropriate and acceptable. The results have been incorporated into the PFP FSAR (1994).

Discussion. The hazards analysis of glovebox MT-5 operations identified as a potential accident the deflagration of flammable gases produced from the thermal decomposition of heavy-metal nitrate-TBP complexes at elevated temperatures. The MT-5 glovebox is used primarily to convert plutonium-bearing residues to nitrate solution by dissolution in near-boiling nitric acid (12M HNO_3) and hydrofluoric acid (0.35M HF). The organic TBP is used in the solvent extraction process at the plutonium reclamation facility and is typically present

in plant recycle feed processed in MT-5. At elevated temperatures (from 130 to 150°C), and in the presence of HNO₃ or heavy-metal nitrates, TBP can exothermally decompose resulting in the rapid evolution of combustible gases, primarily butene. If the generation of flammable gases is significant, an ignition source may cause an explosion that could potentially breach the MT-5 glovebox thus dispersing plutonium-bearing materials. This scenario was analyzed by WHC in WHC-SD-CP-SAR-012 Rev 2 which has been incorporated into the latest revision to the PFP FSAR.

The probability of a deflagration of combustible gases in MT-5 resulting from the decomposition of TBP during normal operations was shown to have an annual event frequency of 10⁻¹⁵/yr. The redundancy built into the dissolver pot controller system significantly contributes to the resultant probability. The probabilistic analysis was reviewed, and while some of the individual probability values used in the analysis may be challenged, it was concluded that the event would still be within the incredible range.

Although the deflagration of flammable gases in MT-5 during normal operations was determined to be an incredible accident scenario, a common cause failure (e.g., a seismic event) may disable many of the safety systems utilized in glovebox MT-5 and, therefore, WHC performed a safety analysis involving a seismically-induced deflagration in glovebox MT-5 (refer to SER Section 9.2.4B). A deterministic analysis of the seismically-induced deflagration, conservatively demonstrates, that the maximum rate at which flammable gases could be generated in glovebox MT-5 is not sufficient to cause a large scale explosion. The safety analysis demonstrates that the consequences associated with a worst-case deflagration in glovebox MT-5 would be small (refer to SER Section 9.2.4B for more details). The DOE review effort determined that the deterministic analysis presented by WHC for the seismically-induced deflagration scenario conservatively bounds the radiological consequences of this event. The scenario assumes failure of the dissolver pot controllers and demonstrates that the consequences associated with this event are acceptable based on WHC's risk acceptance guidelines.

Evaluation Findings. The DOE review concludes that the consequences associated with the seismically-induced MT-5 deflagration analysis (evaluated in Section 9.2.4B) bounds a possible deflagration of combustible gases in the MT-5 glovebox during normal operations. Furthermore, the review concludes that the probability of a deflagration occurring in MT-5 during normal operations is incredible and that the risks associated with such an event are acceptable (see SER Section 9.2.4B).

The following LCOs and ACs represent the minimum OSRs necessary to ensure that glovebox MT-5 operations are conducted within the analyzed safety envelope:

LCO 3.2.1, *Filtered Exhaust From Zones 3 and 4*, specifies the minimum number of HEPA filter stages that must be operable in the ventilation exhaust systems. LCO 3.2.1 also specifies surveillance requirements to test and verify that HEPA filter particle removal efficiencies are at least 99.95%.

LCO 3.3.1, *Control of DISPERSIBLE Plutonium*, and AC 5.22, *Plutonium Inventory*, establishes inventory controls to ensure that facility inventory is equal to or less than the inventory assumed in the analysis.

AC 5.20, *Fire Protection*, establishes the requirements for implementing a fire protection program and identifies the key elements associated with such a program.

Those SSCs listed in Table 4.3 that specifically provide preventive and/or mitigative features for the MT-5 deflagration analysis are Building Structural Features and Final Stage HEPA Filters.

9.2.1D Laboratory Waste Concentrator

Currently, there are no production plans or operational reasons to start up the laboratory waste concentrator and, therefore, this accident scenario has been omitted from the PFP FSAR. If the plant's operational mission changes such that it is necessary to operate the laboratory waste concentrator, WHC shall perform a safety assessment and submit it to DOE for review.

9.2.1E Explosion, Hydrogen in Fluorinator

Currently, there are no production plans or operational reasons to use hydrogen fluoride in the facility and, therefore, this accident scenario has been omitted from the PFP FSAR. Furthermore, WHC has stated that hydrogen fluoride gas is no longer stored within the facility. If the plant's operational mission changes such that it is necessary to use hydrogen fluoride in plant processes, WHC shall perform a safety assessment and submit it to DOE for review.

9.2.1F Flammable Gas Explosion

The acetylene explosion analysis presented in the PFP FSAR (Draft 1991) was found to be technically inadequate. Westinghouse Hanford Company management decided that they would eliminate the use of 75 ft³ and 353 ft³ acetylene gas containers. One of nine open issues requested that WHC make changes to their OSRs to eliminate the use of acetylene gas at the facility and to limit other flammable gases stored and used within the facility. In response to this open issue, a safety assessment of small liquid propane gas (LPG) containers (still required for maintenance activities) was performed by WHC to provide a safety basis for establishing controls. After several internal reviews by WHC's SEAC, and informal DOE reviews, WHC issued WHC-SD-CP-SAD-003 Rev 0, *Safety Assessment of Flammable Gas Explosion at the Plutonium Finishing Plant*, in May 1994. The results of WHC-SD-CP-SAD-003 Rev 0 have been incorporated into the PFP FSAR (1994).

Discussion. The preliminary hazards analysis identified the possibility of a flammable gas explosion at PFP. An accidental mishap involving large acetylene containers could cause a release of flammable gas, and possibly a deflagration and/or detonation. The resulting energy

may result in a breach of confinement and uncontrolled spread of radioactive material. Because of this safety concern, large acetylene gas containers are no longer authorized for use at PFP. However, small, commercially available LPG containers (16.4 oz) are used throughout PFP for performing various activities. The flammable gas explosion safety analysis considers these small LPG bottles and develops administrative controls to control their use.

Liquid propane gas cylinders used at PFP meet the requirements established by the U.S. Department of Transportation and the Compressed Gas Association. In addition, the LPG cylinders are UL listed, having been tested in accordance with UL 147A, *Nonrefillable (Disposal) Type Fuel Gas Cylinder Assemblies*.

Westinghouse Hanford Company considered two release scenarios. The first scenario is a slow leak through normal discharge orifices. The slow leak analysis showed that the worst case was an uncontrolled discharge of propane from a 2.2 mm orifice. Assuming that an ignition source was present the analysis demonstrated that a deflagration in an unventilated 30 m³ room would result in a peak pressure of 0.34 lbf/in². Building wall testing conducted in 1989 shows that the PFP building walls will survive this pressure. The second scenario is a catastrophic failure of the cylinder. The analysis demonstrates that a catastrophic failure of a small LPG cylinder is incredible.

As a result of the flammable gas analysis, WHC identified several administrative controls. The controls are necessary so that facility operations involving LPG cylinders are bounded by the derived safety envelope. Administrative controls specify the quantity of cylinders that may be stored in any one location and the use of LPG within gloveboxes.

Evaluation Findings. The DOE review concludes that the flammable gas safety analysis is acceptable. The preliminary hazards analysis was performed in sufficient detail to identify the most likely failure modes for LPG cylinders and the resultant accident safety analysis demonstrate that the risks associated with small LPG cylinders are acceptable.

The following AC represents the minimum OSRs necessary to ensure that facility operations involving flammable gases are conducted within the analyzed safety envelope:

AC 5.21, *Flammable Gas Inventory*, establishes a program to control the storage and use of flammable gases within the facility.

9.2.1G Explosion, Slag and Crucible System Deentrainer

Currently, there are no production plans or operational reasons to use the slag and crucible process and, therefore, this accident scenario has been omitted from the PFP FSAR. If the plant's operational mission changes such that it is necessary to use the slag and crucible process, WHC shall perform a safety assessment and submit it to DOE for review.

9.2.2 Fire

The review of the facility's fire safety included plant tours and review of the relevant sections of the FSAR including Sections 9.2, 4.3.6, and 5.4.11. Referenced and supporting documentation were not included in the review, except in the area of analysis of the effects of postulated explosions. The depth of the review was based on a graded approach, in that those scenarios that appeared to have little potential for adverse consequences were not reviewed in detail.

The general approach taken by WHC in fire safety has been to reduce the combustible loading in the facility and eliminate, to the extent possible, processes that require highly reactive or toxic chemicals. Westinghouse Hanford Company keeps track of the fire loading in the various facilities that comprise the PFP by performing fire protection surveys (Taylor 1989). The fire loading in the facility is also established in SD-EN-RA-001, *Fire Hazard Analysis for the Plutonium Finishing Plant* and WHC-SD-631-TI-001, *Plutonium Finishing Plant Fire Risk Analysis*. The facilities are constructed out of nonflammable material (concrete and steel) and have been classified as fire resistive/noncombustible. Fire retardant plastic and treated wood are used when these materials are required. To further reduce the fire loading at PFP, WHC has promulgated the following safety requirements:

- Nonflammable hydraulic, lubricating, cooling, etc., fluids shall be used in plutonium handling areas.
- Physical barriers for isolation between incompatible chemicals, materials, and processes shall be provided.

Westinghouse Hanford Company has also stated in the FSAR that routine inspections are to be made each shift to identify and remove trash accumulations. Facility procedures require that combustibles be removed from gloveboxes as they are generated and that weld permits address the necessity of good housekeeping. This approach places less reliance on the fire detection and sprinkler systems, which are not considered safety class systems and are not credited in the fire safety analysis.

The production of plutonium metal has been discontinued, therefore, the use of calcium metal and gaseous HF have been discontinued. The elimination of these materials greatly reduces fire hazards within the facility. The design and good housekeeping requirements are credited with assuring small fire loadings such that fires would not propagate beyond the initiating room and exit filters. The exit filters from rooms and gloveboxes are assumed to fail, but the final stage of HEPA filters are assumed to remain intact.

9.2.2A Remote Mechanical C Line Fire

Discussion. A fire scenario in the RMC line was developed by WHC and fully analyzed. It was considered to be the worst case bounding fire scenario. The scenario is developed based

on conservative assumptions (e.g., the Halon discharge does not extinguish the fire, the Hanford Fire Department (HFD) fails to respond in a normal time, maximum plutonium inventories are in the RMC line and affected systems, and the fire sprinkler system does not activate). The fire is assumed to spread to glovebox HC-18BS where 2000 g of Pu metal buttons are ignited (the actual quantity of Pu metal will be significantly less) and to glovebox HC-11 containing 32,000 g of PuF₄. The fire is finally suppressed by the HFD and the airborne release is mitigated by the HEPA filter banks.

The probability of an RMC line fire was determined to be 1.6×10^{-10} /yr. The radiological consequences resulting from a fire in the RMC line were presented to define a safety envelope to bound all potential fire scenarios at the facility. The resulting radiological consequences were conservatively estimated to be 3.6 rem EDE to the onsite MI and 1.3 rem EDE to the offsite MI.

Evaluation Findings. A review of the RMC line scenario was performed. Several initiating events which may lead to a fire were identified. Westinghouse Hanford Company determined that the fire scenario was incredible, however, the radiological consequences associated with an RMC line fire were presented to bound all other fire scenarios. The DOE review effort concentrated on initial conditions, major assumptions, and radiological dose calculations.

Several conservative assumptions are made by WHC in performing their analysis. All fire suppression systems are assumed to not function correctly, including sprinklers and Halon systems. The HFD fails to respond in what would be considered a normal time. The analysis assumes that the fire spreads to glovebox HC-18BS where 2000 g of Pu metal buttons are ignited (a conservative assumption since metal buttons are no longer stored in the RMC line, although metal buttons may still be stored in glovebox HC-18BS). The maximum number of plutonium powder pans (12 pans, class Y material) are assumed to be involved in the fire. A single stage of HEPA filtration is in place with a 99.95% removal efficiency. Atmospheric dispersion coefficients were based on nine year averaged Hanford site specific joint frequency meteorology data to calculate 99.5% worst case $\bar{\chi}/Q'$ values. Also, the material released into the atmosphere is characterized by a worst case unit inhalation dose associated with the 90th percentile, >10% Pu-240, Class W material.

Westinghouse Hanford Company's analysis of the integrity of the HEPA filter banks during a fire relies on two major assumptions. First, that the ventilation flow continues and causes dilution of the hot combustion gases. And second, that heat transfer occurs through the ventilation duct walls. These assumptions were used to assert that the temperature of the mixed combustion gases and ventilation flow at the inlet to the HEPA filter banks was low enough not to threaten their integrity.

In addition, WHC generally assumed that a fire would be limited to a single compartment. A seismic event has the potential for starting fires in multiple locations and may cause loss of electrical power with concomitant loss of ventilation flow, this raised several issues. With loss of ventilation flow, dilution of combustion gases by airflows from compartments not

involved in the fire may not be a significant mitigating factor. Also, the question of natural circulation flows (in either the usual or reverse directions), driven by fires in one or more compartments, needs to be considered. Westinghouse Hanford Company responded to these issues in Appendix A, Section A.5, questions 1, 2, and 3. They determined that without positive, vigorous air flow, (factors that increase fire severity) the effects on the filters are reduced. Air flow is the primary motive force for the spread of fire to other parts of the glovebox system. Without this air flow, the spread of fire would be less and the glovebox exit filters would limit the amount of reverse flow due to filter plugging from smoke loading. Also, with ventilation failure, the Halon concentration would be high enough to suppress most fires. In addition, the duct volume and distance between the PFP rooms and the final HEPA filters is very large, allowing the combustion gases to cool prior to reaching the filters. The reasoning presented in WHC's responses is found to be acceptable and all abnormal ventilation flow concerns have been sufficiently addressed.

The radiological consequences associated with the bounding fire scenario (RMC line fire) were conservatively calculated to be 3.6 rem EDE onsite and 1.3 rem EDE offsite. Based on the overall conservative assumptions used in the analysis and a comparison with WHC's risk acceptance guidelines (refer to SER Section 9.0.2.2), the DOE review concludes that the bounding fire consequence analysis is acceptable.

The following LCOs and ACs represent the minimum OSRs necessary to ensure that facility operations are conducted within the analyzed safety envelope:

LCO 3.2.1, *Filtered Exhaust From Zones 3 and 4*, specifies the minimum number of HEPA filter stages that must be operable in the ventilation exhaust systems. LCO 3.2.1 also specifies surveillance requirements to test and verify that HEPA filter particle removal efficiencies are at least 99.95%.

LCO 3.3.1, *Control of DISPERSIBLE Plutonium*, and AC 5.22, *Plutonium Inventory*, establishes inventory controls to ensure that facility inventory is equal to or less than the inventory assumed in the analysis.

AC 5.20, *Fire Protection*, establishes the requirements for implementing a fire protection program and identifies the key elements associated with such a program.

Those SSCs listed in Table 4.3 that specifically provide preventive and/or mitigative features for the RMC Line Fire analysis are Building Structural Features and Final Stage HEPA Filters.

Although Pu metal will no longer be produced in the facility, the RMC line fire scenario is assumed to bound all other fire scenarios because the amount of plutonium considered in the analysis is very conservative. Therefore, the dose consequences are found to be both conservative and acceptable. To the extent practical, the facility and management practices relating to fire safety meet DOE objectives. Given the limited mission of the PFP facility, the

DOE review concludes that WHC's fire analysis is adequate and that sufficient OSRs and ACs exist to ensure that facility operations are conducted within the analyzed safety envelope.

9.2.2B Roof Fire, 234-5Z

Discussion. The probability of a fire occurring on the roof of the 234-5Z building and causing a plutonium release from the filter rooms was identified by WHC and included in the PFP FSAR. The scenario was shown to be an incredible event, and the radiological consequences were not calculated.

Evaluation Findings. The RMC line fire scenario was determined to be the bounding fire scenario and, therefore, a detailed review of the postulated roof fire scenario was not performed. The DOE review concludes that sufficient OSRs exist to ensure that facility operations are conducted in such a manner that the likelihood of a fire occurring at PFP is low. Furthermore, DOE concludes that WHC has sufficiently developed the resulting consequences of their bounding fire scenario so as to understand the radiological risks associated with a fire at the facility.

9.2.2C Glovebox MT-5 Fire

The original MT-5 safety analysis presented in the PFP FSAR (Draft 1991) was found to be unacceptable. One of the nine open issues was related to the MT-5 accident scenario. The initial safety analysis concluded that a seismically qualified MT-5 hot-plate power cut-off switch should be installed to ensure that the power to the MT-5 dissolvers would be secured in the event of an earthquake. The switch was never installed and, therefore, the safety analysis was considered invalid. As a result, WHC released a draft of WHC-SD-CP-SAR-012 Rev 2, *MT-5 Safety Assessment* (October 1993). This safety assessment included a more detailed and rigorous analysis of the chemical processes occurring in the MT-5 dissolver pots. The analysis incorporated new laboratory studies which provided experimentally determined reaction rates relating to the chemistry in the MT-5 dissolver pots. After several internal reviews by WHC's SEAC, and informal DOE reviews, a final MT-5 safety assessment document was issued in June 1994. The DOE reviewers concluded that the safety assessment presented in WHC-SD-CP-SAR-012 Rev 2 was appropriate and acceptable. The results have been incorporated into the PFP FSAR (1994).

Discussion. To determine the risk associated with a fire in glovebox MT-5, WHC performed a deterministic radiological dose consequence analysis assuming a maximum plutonium inventory and an ignition source. An ignition source is considered to be unlikely by WHC but for added conservatism this was assumed to occur in their safety analysis. The analysis assumes a fire occurs in glovebox MT-5 and a maximum quantity of plutonium is consumed in the fire. The aerosol resulting from plutonium consumed in a fire is transported out of the room through both the E-4 and E-3 ventilation systems. The material is then assumed to pass through one stage of HEPA filtration before it is released out of the 291-Z-1 stack.

The analysis shows a maximum radiological dose of approximately 0.8 mrem EDE is received by the onsite MI and approximately 0.3 mrem EDE is received by the offsite MI. Westinghouse Hanford Company's risk acceptance guidelines show that the risks associated with an MT-5 glovebox fire are acceptable based on an anticipated event.

Evaluation Findings. The DOE review effort concludes that the initial conditions stated in the analysis and that the plutonium inventory assumed in the analysis are both conservative. Furthermore, the 90th percentile >10% Pu-240 unit inhalation dose conservatively describes the material that may be present in glovebox MT-5 during plant operations, and that bounding respirable release fractions were used in the analysis. Based on the resultant radiological dose consequences, and a comparison with WHC's risk acceptance guidelines, the DOE review concludes that the risks associated with a fire in MT-5 are acceptably low.

The following LCOs and ACs represent the minimum OSRs necessary to ensure that glovebox MT-5 operations are conducted within the analyzed safety envelope:

LCO 3.2.1, *Filtered Exhaust From Zones 3 and 4*, specifies the minimum number of HEPA filter stages that must be operable in the ventilation exhaust systems. LCO 3.2.1 also specifies surveillance requirements to test and verify that HEPA filter particle removal efficiencies are at least 99.95%.

LCO 3.3.1, *Control of DISPERSIBLE Plutonium*, and AC 5.22, *Plutonium Inventory*, establishes inventory controls to ensure that facility inventory is equal to or less than the inventory assumed in the analysis.

AC 5.20, *Fire Protection*, establishes the requirements for implementing a fire protection program and identifies the key elements associated with such a program.

Those SSCs listed in Table 4.3 that specifically provide preventive and/or mitigative features for the Fire in Glovebox MT-5 analysis are Building Structural Features and Final Stage HEPA Filters.

9.2.2D Waste Drum Fire

After reviewing the various accidents identified by WHC in the PFP FSAR, an issue was raised by DOE regarding the potential hazards associated with the storing of 55 gallon drums containing radioactive waste outside the facility. After some consideration, WHC concluded that such a scenario was indeed plausible at PFP. Westinghouse Hanford Company considered three different initiating events that could adversely affect waste drums stored outside the facility: (1) a vehicle crash causing 200 radioactive waste storage drums to be consumed by fire; (2) a mechanical release due to a waste drum being overpressurized or ruptured by some external means; and (3) a single drum explosion resulting in a fire. The safety analysis establishes the amount of plutonium-equivalent transuranic waste that can be stored within the fenced security area outside the facility.

Discussion. The most limiting accident scenario involving 55-gallon drums containing contaminated waste was shown to be a fire involving 200 55-gallon drums temporarily stored outside the facility awaiting transfer to a permanent storage site. The initiating event was postulated to be a vehicle crashing into the stored waste drums resulting in a fire that is conservatively assumed to consume the contents of all 200 drums. The source term is limited to a total of 3000 grams of transuranic (TRU) material, each drum being limited to 15 grams in accordance with criticality safety control limits. The respirable fractions and airborne release fractions used in the analysis were reviewed against data presented in Mishima (1993) and found to be acceptable. The 99.5% worst-case atmospheric dispersion coefficient credits the effects of a buoyant plume rise resulting from the fire (see SER Section 9.0.5.1). Westinghouse Hanford Company provided adequate justification for using a >10% Pu-240 mass weighted average unit inhalation dose conversion factor to characterize the source term. The deterministic radiological dose consequence analysis results in 2.2 rem EDE to the onsite MI and 0.04 rem EDE to the offsite MI. A detailed probabilistic analysis was not performed, instead WHC assumed the event to be unlikely (i.e., 10^{-2} to 10^{-4} /yr).

A mechanical release scenario involving the rupture of a 55-gallon waste drum caused by overpressurization, vehicle impact, or a forklift handling mishap was also analyzed and reviewed. The respirable fractions and airborne release fractions used in the analysis were reviewed against data presented in Mishima (1993) and found to be acceptable and bounding. The atmospheric dispersion coefficient, based on a ground level point release Gaussian plume model, represents the 99.5% worst-case dispersion and is conservative. The 90th percentile >10% Pu-240 unit inhalation dose conversion factor used to characterize the source term material will adequately bound the material stored in a 55-gallon waste drum. The analysis conservatively results in 0.12 rem EDE to the onsite MI and 0.002 rem EDE to the offsite MI. A detailed probabilistic analysis was not performed; WHC simply assumed a worst case frequency of 10^{-1} /yr and demonstrated that the dose consequences were below their risk acceptance guidelines.

An explosion scenario in a single waste drum causing a fire was also developed and analyzed by WHC. The respirable fractions and airborne release fractions used in the analysis were reviewed against data presented in Mishima (1993) and found to be acceptable. The 99.5% worst-case atmospheric dispersion coefficient credits the effects of a buoyant plume rise resulting from the fire (see SER Section 9.0.5.1). The 90th percentile >10% Pu-240 unit inhalation dose conversion factor used to characterize the source term material will adequately bound the material stored in a 55-gallon waste drum. The deterministic radiological dose consequence analysis results in 1.65 rem EDE to the onsite MI and 0.02 rem EDE to the offsite MI. A detailed probabilistic analysis was not performed, instead WHC considered the event to be unlikely (i.e., 10^{-2} to 10^{-4} /yr).

Evaluation. The DOE review concludes that the risks associated with the storage of waste drums outside the facility are acceptable. Westinghouse Hanford Company has conservatively demonstrated that the radiological dose consequences associated with accidents involving waste drums stored outside the facility are within their risk acceptance guidelines.

The following AC represents the minimum OSR necessary to ensure that the storage of waste drums outside of PFP is conducted within the analyzed safety envelope:

AC 5.24, *Management of Transuranic Material Outside Facility Buildings*, establishes specific controls on waste drum container specifications, the amount of radioactive material that may be loaded into a single drum, and the total amount of transuranic material that may be stored in drums outside the facility.

9.2.3 Criticality

The following FSAR sections were specifically reviewed: 4.3.4, Nuclear Criticality Safety; 6.1.3.1, Criticality Prevention; 9.2.3, Criticality; 9.2.4, Seismic Event; 10.1.2, Plant Operating Organization; 10.3, Training Programs; 11.2.1 Criticality Prevention; 11.3.1, Criticality Prevention System; 11.3.2, Criticality Detectors and Alarms; 11.5, Administrative Controls. Nuclear criticality safety was reviewed against the requirements in DOE Order 6430.1A, DOE Order 5480.5 with guidance from DOE Order 5480.23, and ANS/ANSI standard 8.1.

As part of the review, several questions were formally asked of the WHC staff and were responded to by WHC staff (refer to Appendix A). Many of the questions simply requested additional information and are not addressed in this written review. Only those questions that were resolved with complex or less-than-ideal answers are discussed in this evaluation.

Westinghouse Hanford Company has taken a three-step approach to analyzing criticality accidents. First, to demonstrate that criticality is unlikely during normal operations or a seismic event; second, to assume that a criticality accident occurs; and third, to show that if a criticality does occur, its consequences are acceptable. This approach is satisfactory.

9.2.3.1 Recovery (FSAR 9.2.3.3)

Discussion. No specific criticality recovery plan was presented. However, WHC indicated that from past experience, the recovery would be slow and deliberate with formal written procedures that would be reviewed by DOE. The criticality would be terminated by physical phenomena resulting from the criticality. Criticality in metal systems would stop after metal melting and relocation; powder systems would stop by material expansion and scattering; and aqueous solutions would be terminated by microbubble formation and liquid evaporation.

Evaluation Findings. The approach proposed by WHC for the PFP facility is acceptable based on DOE Orders 5480.5 and 6430.1A. Criticalities usually occur during unique unplanned conditions and terminate by physical changes caused by the criticality. The result is that once a criticality occurs and then terminates, careful analysis, planning, and review is an appropriate and preferred approach. The uniqueness of any criticality accident makes it extremely difficult to develop a detailed recovery procedure before the event.

9.2.3.2 Accident Analysis (FSAR 9.2.3.5)

Discussion. It was assumed that if a criticality accident occurs, the criticality would be bounded by historical criticality events (accidents or experiments). Westinghouse Hanford Company reviewed past criticality accidents and conservatively estimated that their assumed accident would have more fissions than any previous event. The magnitude of the criticality, the number of fissions, is identical to the value prescribed by the NUREG-3.35 with an initial burst and several smaller consequent bursts totaling 1.0×10^{19} fissions. This number is an upper bound for the number of fissions occurring in a large liquid accidental criticality; other systems would likely have a fewer total number of fissions. Criticality during a seismic event is discussed in FSAR Section 9.2.4A.2.11.

Westinghouse Hanford Company analyzed a generic bounding criticality accident to determine the criticality safety envelope. The criticality accident was assumed without identifying specific accident scenarios or initiating events. The criticality analysis performed by WHC used the following assumptions:

- all noble gas fission products and 25% of the iodine radionuclides are released in the accident, and are released directly to a ventilated room with no holdup during the accident,
- criticality occurs in solution and 0.05% of the fission products and plutonium are released to the air during the accident,
- radionuclides are not held up in the duct work,
- release to the environment is through a single HEPA filter,
- transmission factor for a single stage of HEPA filters is 5.0×10^{-4} ,
- HEPA filters do not filter fission product gases or radioactive iodine,
- and people onsite are evacuated following the criticality accident.

The criticality controls make an accident unlikely during operation. General scenarios were developed identifying the requirements for metal systems, powder systems, general small liquid systems, and the waste liquid systems. Each of the scenarios was determined to be unlikely. However, no specific accident chains and probabilities were produced that demonstrated that a criticality accident is incredible. Instead, analysis of known criticality accidents and the number of DOE sites involved with those accidents was presented to suggest that criticality accidents occur less than one per 40 site-years of operation. This analysis suggested that, based on past history, the probability of a criticality accident is decreasing.

The resulting radiological exposure associated with a criticality was determined for two release paths; a release from the 291-Z-1 stack, and a ground level release. Atmospheric dispersion coefficients were based on nine year averaged Hanford site specific joint frequency meteorology data to calculate 99.5% worst case $\bar{\chi}/Q'$ values. The stack release analysis results in a 0.16 rem EDE exposure to the onsite MI and 0.032 rem EDE for the offsite MI. The ground level release results in a 0.73 rem EDE exposure to the onsite MI and 0.053 rem EDE for the offsite MI.

Evaluation Findings. Based on the review of the PFP FSAR and responses to the formal questions, the FSAR adequately covers criticality safety and criticality accidents. The methods proposed should be adequate to prevent criticality accidents during normal operations. Specific application of the methods or computer calculations were not in the FSAR and were not reviewed. During normal operations all of the important errors are covered by the error descriptions in the FSAR.

The criticality analysis showing that no criticality will occur during normal operation is acceptable. The argument that the probability of a criticality is lower than once per 40 years is not sufficiently developed and, therefore, is not acceptable. The argument that the probability of a criticality on the order of once in 40 years is reasonable given the difficulty of constructing scenarios that result in criticalities and calculating the probability of a criticality. The FSAR used the probability of a criticality of once per 40 years and the conclusions are acceptable.

The consequence analysis assumed that a criticality accident occurs and demonstrated that the radiological consequences associated with a criticality are below WHC's risk acceptance guidelines (refer to SER Section 9.0.2.2). Westinghouse Hanford Company used the criticality analysis methodology presented in NUREG-3.35 as the basis for analyzing a criticality event.

The following LCOs and ACs represent the minimum OSRs necessary to ensure that facility operations are conducted within the analyzed safety envelope:

LCO 3.1.1, *Criticality Prevention System*, requires the operability of specific criticality safety features at the facility.

LCO 3.1.2, *Criticality Detectors and Alarms*, specifies the operability requirements for facility criticality detectors and alarms.

AC 5.15, *Nuclear Criticality Safety*, establishes the requirements for implementing a nuclear criticality safety program and identifies the key elements associated with such a program.

Those SSCs listed in Table 4.3 that specifically provide preventive and/or mitigative features for the Criticality analysis are Building Structural Features, 26 Inch Vacuum Liquid Detection and Interlock, Glovebox Sump Disks and Floor Filler Plates, HC-6 Raschig Rings & Glovebox Criticality Drains, and Plutonium Storage Arrays & Criticality Alarm Systems.

9.2.4 Seismic Event

The review and evaluation of the seismic event consisted of 6 inter-related steps. The steps are consistent with those used by WHC to develop the seismic event. Step 1 (SER 9.2.4.1)

reviews the seismic qualification of the PFP structures. Step 2 (SER 9.2.4.2) evaluates the postulated seismic induced accident scenarios developed by WHC thus providing an understanding of how the various gloveboxes and facility components were assumed to respond during an seismic event. Step 3 (SER 9.2.4.3) reviews the probability of the design basis earthquake for the Hanford site and its associated ground acceleration. Step 4 (SER 9.2.4.4) reviews and evaluates the method used by WHC to determine the quantity of material released into the facility based on both the documented seismic qualifications of facility structures and gloveboxes, and the postulated accident scenarios. Step 5 (SER 9.2.4.5) reviews the method used by WHC to determine the fraction of material that is released out of the facility and into the environment. And step 6 (SER 9.2.4.6) reviews the radiological dose consequence calculations for the seismic event. (Please note that the enumeration of subsections in 9.2.4 of this report is different than those in the FSAR, therefore the appropriate FSAR section has been provided in parentheses.)

9.2.4A.1 Seismic Qualification of PFP Structures

(see FSAR Section 9.2.4A *Bulk of Facility*)

234-5Z Building and Panel Methodology

Discussion. The original 234-5Z building seismic analyses (Blume 1980) were historically conservative and indicated that the 0.25 g event would produce a response in the building that would breach confinement. Subsequent to that finding, URS/Blume (Blume 1987) undertook the task of performing a nonlinear direct integration time-history analysis for the facility. The building was designated as a High Hazard building and evaluated for 0.25 g zpa. The analysis was done in order to get a more accurate assessment of the structure's nonlinear capacity and to eliminate sources of conservatism in the original analysis. The results of the nonlinear seismic analysis of the 234-5Z building showed that there were a few structural members within the facility that experienced some material yielding, but in general the building remained elastic. The results also showed that the structure did not breach confinement as the original, conservative analysis indicated. The nonlinear analysis also indicated that some of the side panels would tend to tear out and buckle which could possibly provide a pathway for release. Subsequently, WHC initiated a set of documented panel tests which addressed the potential damage to panels during the seismic event (Giller 1989). The results of those shear tests revealed that the panels were more rigid than the nonlinear analysis showed. The largest effect of the seismic loads was to elongate bolt holes where the panels were attached.

Evaluation Findings. The original URS/Blume (Blume 1980) analysis showed that the 234-5Z building would breach confinement during the seismic event. That result was based upon modal analyses. When WHC determined that the potential breach of confinement of 234-5Z was unacceptable, URS/Blume performed the nonlinear dynamic analysis using the direct integration finite element approach. UCRL-15910 (Kennedy 1990) recommended this approach if it was required to reduce conservatism and show that the facility would not breach confinement. An important aspect of the nonlinear analysis was obtaining a time-history that

adequately and conservatively matched the response spectrum. A question was raised (refer to Appendix A, Section Q-8) regarding the possibility that the time-history curve used was not the worst case curve that matched the response spectrum. The answer to that question was provided in Murry (1988) and centered on the fact that the nonlinear direct integration analysis demonstrates that only several of the structural members of the facility experienced significant yielding as opposed to general yielding in most of the structural members. If significant nonlinear response had occurred, then there would have been additional work required to determine whether the time-history curve used was applicable. Furthermore, the analysis was performed for a 0.25 g event rather than the currently required 0.20 g event for High Hazard facilities on the Hanford reservation (SDC 4.1). The DOE review concludes that the seismic evaluation of the 234-5Z Building meets the DOE requirements for a High Hazard building as specified in UCRL-15910.

236-Z Building Methodology

Discussion. Westinghouse Hanford Company classified the 236-Z Building as a High Hazard facility and used the modal superposition method and the Hanford site seismic response spectrum anchored to 0.25 g to perform the seismic evaluation. The results of the analysis (Blume 1988) indicated that the building was capable of resisting seismic loads with the following two exceptions. An east-west wall had a demand/capacity ratio of 1.23 against an overturning moment, but an additional east-west wall near the overstressed wall had adequate excess load capacity to carry the additional loading from the overstressed wall. The analysis also indicated that a concrete block wall was found to have a demand/capacity ratio greater than 1.0, but it was not a load carrying wall and did not change the 236-Z Building's capability to resist the seismic loading. However, failure of this wall would compromise the confinement safety function, for both the DBE and an explosion within a process vessel (this scenario has been removed from the FSAR since the vessel will not be operated). It was primarily the loading from the vessel explosion that governed the design of modifications to the block walls. These modifications consisted of welding stiffened steel plates to structural steel members embedded in the surrounding reinforced concrete walls. Westinghouse Hanford Company also performed a finite element analysis of the 12 ft. by 11 ft. south wall door for seismic and pressure loadings from the glovebox explosion to determine if confinement was maintained. The analysis concluded that studs connecting the two door segments at the centerline were overstressed. Recently, additional analyses were performed to evaluate the behavior of the door without the connecting studs. This analysis indicated that a maximum lateral deflection of 2 inches at the centerline of the 12 ft. by 11 ft. door opening could occur during a seismic event while under elastic conditions, i.e. all stresses remained below the yield value. Therefore, the 2 inch deflection only occurs for the duration of the seismic event.

Evaluation Findings. The analysis for the 236-Z building was performed using a dynamic analysis as required in UCRL-15910 for High Hazard facilities. The response spectrum was based on a Blume study performed in 1988 for the Hanford site and anchored to a zpa of 0.25 g. This zpa exceeds the current design basis value of 0.20 g in SDC 4.1 and is therefore considered satisfactory. The design response spectrum is also considered satisfactory since it

was derived using UCRL-15910 methodology. Two areas of the building had a demand/capacity ratio greater than 1.0 as discussed previously. The east-west wall with the D/C ratio of 1.23 was judged satisfactory by WHC on the basis of the excess capacity of a nearby wall. This assessment was made based on engineering judgement without any specific analysis. The DOE considers this approach less than satisfactory but accepts the D/C ratio of 1.23 because the analysis was for a zpa of 0.25 g instead of the current 0.20 g zpa design value (for a 0.20 g D/C ratio is 0.98). Westinghouse Hanford Company has also shown that the stiffened steel plates referred to previously can withstand the amplified inertial forces of the block wall during a seismic event. It was shown that these inertial forces were equivalent to a uniform 1.8 psi lateral pressure while the pressure loading from the explosion was 2 psi. Although the steel plates preclude the walls from collapsing outward, the unreinforced block wall may collapse inward during the DBE. The possibility of the upper portion of the block wall collapsing inward was considered during the review process.

The DOE review determined that there were no Safety Class SSC's which might be impacted by the falling concrete blocks. The DOE discussed with WHC the possibility of the concrete blocks falling and disturbing contamination in the form of sludge on the floor of the process cell, thus causing an additional airborne release. The DOE review concluded that such a concern was insignificant based on the quantity of sludge that may be affected and the relatively small respirable release fractions associated with such forms of material.

The DOE also believes the 2 inch lateral separation of the two door segments does not pose a significant hazard because of the short duration of the seismic event and the high probability that the negative pressure differential will not be lost for this brief time period. The basis for the pressure differential is derived from the seismically triggered shutoff of the supply fans and the expected coast-down time for the exhaust fans in the event power is lost. Therefore, the ability of the doors to confine material released inside the process cell is maintained throughout a seismic event. On the basis of the above discussion DOE concludes that the performance of building 236-Z is adequate for the seismic event and poses no significant hazard to the safety of the public and the workers.

242-Z Building Methodology

Discussion. The 242-Z Building was classified as a High Hazard and evaluated for a 0.25 g zpa seismic event. The FSAR indicated that the 242-Z building was analyzed using modal analysis (Giller 1988). Through the use of modal expansion techniques individual member deformation and stresses were obtained. The analysis demonstrates that during a seismic event the building will suffer insignificant damage with the exception of an overstress condition in two anchor bolts at the base of one column and a 20 percent overstress condition in two cross bracing rods. Westinghouse Hanford Company's recommendations were to replace the cross braces and modify the base plates prior to plutonium nitrate storage activities.

Evaluation Findings. A modal analysis was used to seismically evaluate the 242-Z Building. A question concerning the status of the cross bracing replacement and base plate modifications in preparation for plutonium nitrate solution storage was posed to WHC. The WHC response was that 242-Z was no longer being considered for storage of plutonium nitrate, so the upgrade to 242-Z was no longer required. The plan for upgrading the building is no longer required since 242-Z will not be used for plutonium nitrate solution storage. Based upon the new PFP mission and operations, the WHC conclusion concerning the upgrades is acceptable. The DOE review concludes that the seismic evaluation of the 242-Z Building meets the requirements for a High Hazard building as specified in UCRL-15910.

291-Z Stack Methodology

Discussion. In 1980 URS/Blume performed a seismic evaluation of the 291-Z stack (Blume 1980) that showed the stack was overstressed and the 291-Z Building would not suffer significant damage during a seismic event. This analysis did not account for the soil structure interaction (SSI) effects. The inclusion of SSI could reduce the effects of the seismic loading at the base of the structure.

The 291-Z stack was designated a High Hazard structure and evaluated for a 0.25 g zpa. In 1988, the 291-Z stack was analyzed (Blume 1988) using two approaches. The first, a 2-D "stick" model was developed and the SSI effects were addressed using frequency dependent springs to model the soil half space. The spring properties were based upon SSI using the FLUSH code for the 291-Z stack and 291-Z building. In the second approach URS/Blume created a three dimensional nonlinear model of the stack and performed a time-step direct integration analysis. The three dimensional stack analysis indicated that there were areas of concrete cracking and plastic deformation in some of the reinforcement. The results from the nonlinear analysis were then introduced as material property changes in the stick model. These two models were then compared to each other to determine how severe the nonlinearities were and also to help verify the analyses. Although there would be some areas of concrete cracking and steel reinforcement yielding, the 1988 analyses results indicated that the stack would not suffer significant damage and will continue to perform its function.

Evaluation Findings. The analysis performed by URS/Blume in 1980 showed that the stack was overstressed. This analysis did not indicate that the stack would suffer significant damage, but it was decided that a more rigorous analysis was needed to demonstrate that the stack was not overstressed. The FSAR did not provide adequate details about the analyses performed by URS/Blume (Blume 1988). The URS/Blume document has been reviewed to examine details of the analyses which are not in the FSAR. The DOE review concludes that the seismic evaluation of the 291-Z stack meets the DOE requirements for a High Hazard structure as specified in UCRL-15910.

291-Z Building Methodology

Discussion. The 291-Z Building was classified as a High Hazard facility and evaluated for a 0.25 g zpa seismic event. A preliminary analysis (Blume 1980) demonstrated that the 291-Z building would be able to resist the seismic event although the stack would be overstressed; however, the analysis did not take into account the soil structure interaction (SSI). A second analysis included both the 291-Z Building and the 291-Z stack and considered SSI effects. This analysis served as a confirmatory analysis for the 291-Z Building. The SSI analysis indicated that the facility would be able to withstand seismic loading with the exception of some yielding of the reinforcement in the first floor beams under the exterior or interior walls. The yielding would cause stress redistribution to the walls which will adequately sustain the additional load.

Evaluation Findings. The summary of the preliminary analysis of the 291-Z Building indicated that the building would have insignificant damage during the seismic event; however, detailed calculations of the analysis were not available for review. Based on the summary of the preliminary analysis, the assumptions and methods used to analyze the building were found to be acceptable. The DOE review concludes that the seismic evaluation of the 291-Z Building meets the requirements for a High Hazard building as specified in UCRL-15910.

241-Z Building Methodology

Discussion. Westinghouse Hanford Company concluded that an airborne release from the 241-Z Building would be negligible (refer to FSAR Section 9.2.4A.2.10) since the cover blocks would withstand the DBE. An earlier analysis (Marusich 1988) had assumed failure of the cover blocks during the seismic event resulting in an airborne release of 0.6 grams Pu. Subsequent to that analysis WHC performed the seismic evaluation of the cover blocks (Ocoma 1988) and hence concluded in the FSAR that the release would be negligible.

Seismic analyses of the 241-Z below-grade reinforced concrete building (Ocoma 1988) were reviewed. The analysis was based on an equivalent static force analysis method and a 0.25-g zpa seismic event. The seismic analysis demonstrated that the reinforced vault cover blocks, which are made of 6 inch thick concrete with 3/4 inch rebar reinforcement, will withstand DBE motions. The seismic capacity of the cell interior walls within the building was calculated and determined to be adequate. During the review a concern was raised regarding the potential interaction effects of an overhead crane and the metal framed building located over the reinforced vault cover blocks (the crane provides a lifting capability for removing the cover blocks). Also, a concern was raised regarding the below grade exterior walls which were assumed to withstand seismic loads based on the evaluation results (i.e. sufficient margins) of the interior walls.

To address the concerns regarding the interaction effects and the qualification of the exterior walls, the DOE review considered the original safety analysis which WHC used to conclude

that the cover blocks would withstand the DBE event (Marusich 1988). The analysis assumed that a cover block would be dislodged from its position and fall onto the tank below. A release was calculated based on a round cover block falling directly into a cylindrical tank and acting as a hydraulic piston forcing 1/2 of the tank's contents (5,000 liters) into the atmosphere. A bounding ARF/AF value of $1E-4/1.0$ was used to estimate the mass fraction of droplets 10 micrometers and less in diameter formed by the spray. This is a bounding value from Mishima (1993) which is based on experimental data derived from a spray nozzle. The ARF/AF value was applied to the maximum plutonium content in all 5 tanks. An order of magnitude error (too high) was made in the source term calculation by a misplaced decimal point. Upon further investigation, it was concluded that a bounding source term released to the atmosphere from all 5 tanks would be 0.05 grams respirable Pu. This represents less than 3 percent of the total source term in the no ventilation case which is a ground level release (see SER section 9.2.4A.13).

Even if the outer walls and the overhead crane were to fail the release would be bounded by that determined above (i.e., one half of the tanks' liquid content are sprayed into the atmosphere), therefore, the 0.05 gram source term would remain applicable. Based on these considerations the concerns raised regarding the interaction effects, and qualification of the below grade exterior walls and overhead crane, were found to have an insignificant effect on the source term even if the exterior walls and crane were to fail.

Evaluation Findings. Westinghouse Hanford Company performed a seismic evaluation of the storage tank pit covers and interior walls located within the 241-Z building. The capacity was determined using the working stress design method. The capacities of the pit cover and the interior walls are much higher than demand. The potential interaction effects of an overhead crane were not considered. The exterior walls of the vault are subjected to forces due to soil pressure and other vertical loads from the structure above. These walls have more reinforcement and are thicker than the interior walls, thus WHC qualitatively argued that they would withstand seismic loads. However, the capacity and demand of these walls and the overhead crane were not determined. An evaluation of the bounding scenario concluded that the source term would be negligible when compared to the source term associated with the seismic event.

2736-Z Building Complex Methodology

In 1993, the DOE review concluded that WHC did not have sufficient documented seismic analyses to defend the seismic qualification of the 2736-Z, ZA, and ZB Buildings. As a result, DOE arranged for a two day meeting to be held October 6th and 7th to discuss the concerns regarding the inadequacy of documentation to demonstrate the seismic qualification of the 2736 complex. Participants in the meeting consisted of staff members from DP-31, PNL, and WHC structural engineers. During the course of the meeting WHC presented to DOE structural drawings of the buildings, seismic design requirements in place at the time the buildings were constructed, and previous seismic reviews. Westinghouse Hanford Company also provided a tour of the 2736 complex for visiting DOE and PNL staff.

Based on the tour of the 2736 complex structures and systems, a review of structural drawings, the past seismic design requirements, and previous associated seismic analyses, it was concluded that there was a high probability that the 2736 structures and systems could be seismically evaluated to UCRL-15910 standards with little or no modifications. Furthermore, it was agreed upon by both WHC and DOE that WHC would perform seismic analyses in accordance with the requirements in UCRL-15910 to support their conclusions in the FSAR that the 2736 complex structures and systems would withstand a DBE.

The following evaluations pertain to the 2736 complex seismic analyses that WHC performed during the period October 1993 to October 1994. Seismic evaluations of the safety class SSCs located within these buildings is ongoing. The current PFP FSAR (1994) references the latest seismic analyses of the structures.

2736-Z Building

Discussion. The DOE review included WHC's seismic evaluation of the 2736-Z structure presented in Giller (1994a). The 2736-Z Building is a single story reinforced concrete plutonium storage vault. Westinghouse Hanford Company classified the 2736-Z structure as a Safety Class 1 structure. Westinghouse Hanford Company used an equivalent static method to evaluate the seismic load on the facility. The analysis resulted in positive margins of safety with respect to material design strengths for a 0.2g ZPA earthquake and damping ratio of 7%. An equivalent static method was used to perform the seismic evaluation based on a provision in Appendix A of UCRL-15910, which states, "...for very important or highly hazardous facilities, such as the Moderate or High Hazard categories, it is recommended that the equivalent static force method not be used except for very simple structures." Westinghouse Hanford Company provided evidence that the 2736-Z building is a simple structure based on provisions in the Uniform Building Code (UBC). During the review process WHC was asked to provide a finite element analysis (FEA) to determine the shear demand for each column in a plane of the building. The FEA indicates a shear demand in one of the 2736-Z building columns equal to 34 kips. The capacity of this column was determined to be 30.9 kips resulting in a demand to capacity (D/C) ratio of 1.15. Westinghouse Hanford Company has provided their position regarding the FEA to DOE and they have concluded that no overstress condition exists for the column. The DOE review does not agree with WHC's position and believes that the 1.15 D/C ratio does constitute an overstressed condition for the column. However, DOE also believes that the argument presented in "evaluation findings" provides an appropriate technical justification for concluding that the 1.15 D/C ratio in the 2736-Z column is an acceptable condition.

The primary confinement system located within the 2736-Z building is comprised of steel racks, cubicles, pedestals, and plutonium containers and has been classified by WHC as a Safety Class 1 system. The seismic evaluation of this system is ongoing. Westinghouse Hanford Company has committed to complete the seismic evaluation of this system and to perform any necessary seismic upgrades that may be identified during the evaluation process (Ruff 1994).

Evaluation Findings. The assumption made by WHC that the structure is a simple structure has been accepted by DOE because:

- torsional irregularities in the 2736-Z Building has less than 5% eccentricity, which is lower than the 10% limit in the UBC code,
- the 2736-Z Building is a single story structures and all have parallel lateral load system,
- diaphragms of the building does not have abrupt variations in stiffness, nor any openings greater than 50% of the gross area,
- the 2736-Z Building used shear wall diaphragms for lateral resistance, and all lateral load members are in line with these diaphragms, and
- with exception of the 2736-Z Building, the 2736-Z complex buildings do not have discontinuities in the lateral load path including out-of-plane members.

The 2736-Z Building lateral load systems were analyzed using the finite element method to address the discontinuities in the load path. Based on these analyses, the columns in the same wall carried unequal shear forces. This distribution of forces among the columns results in an overstressed column. The demand to capacity ratio is 1.15. Since this is an existing facility, DOE believes that a somewhat increased risk is acceptable. The increased acceptable risk corresponds to doubling the hazard annual probability of exceedance. For the increased risk, the demand to capacity ratio is 0.98. The DOE review concludes that the seismic evaluation of the 2736-Z Building is consistent with guidelines provided in UCRL 15910, and is therefore acceptable for the associated risk.

Westinghouse Hanford Company has committed to complete the seismic evaluation of the steel racks, cubicles, and pedestals that were identified as a discrepancy. Since these discrepancies involve a condition in an existing facility that is currently operating, and since resolution of the discrepancy is being aggressively pursued by WHC, DOE has agreed to the schedule of completion as outlined in Ruff (1994).

2736-ZB Building

Discussion. The review of WHC's seismic evaluation of the 2736-ZB structure, documented in Weiner (1994), was completed. The 2736-ZB building provides support functions for the 2736-Z plutonium storage vaults including shipping/receiving functions, a nondestructive plutonium assay laboratory, and a glovebox for repackaging plutonium containers from the 2736-Z vault. Westinghouse Hanford Company classified the 2736-ZB structure as a Safety Class 1 structure. Westinghouse Hanford Company used an equivalent static method to evaluate the seismic load on the facility. The analysis resulted in positive margins of safety with respect to material design strengths for a 0.2 g zpa earthquake and damping ratio of 7%. An equivalent static method was used to perform the seismic evaluation based on a provision in Appendix A of UCRL-15910, which states, "...for very important or highly hazardous facilities, such as the Moderate or High Hazard categories, it is recommended that the equivalent static force method not be used except for very simple structures." Westinghouse Hanford Company provided evidence that the 2736-Z building is a simple structure based on provisions in the Uniform Building Code (UBC).

A repackaging glovebox is located within the 2736-ZB building and provides primary confinement of open containers within the glovebox. Westinghouse Hanford Company has identified the repackaging glovebox as Safety Class 1. A seismic evaluation of the glovebox is ongoing. Westinghouse Hanford Company has committed to complete the seismic evaluation of the glovebox and to perform any necessary seismic upgrades that may be identified during the evaluation (Ruff 1994).

Evaluation Findings. The DOE review concludes that the seismic evaluation of the 2736-ZB Building is consistent with guidelines provided in UCRL 15910, and is therefore acceptable.

Westinghouse Hanford Company has committed to complete the seismic evaluation of the repacking glovebox in the 2736-ZB Building that was identified as a discrepancy. Since this discrepancy involves a condition in an existing facility that is currently operating, and since resolution of this discrepancy is being aggressively pursued by WHC, DOE has agreed to the schedule of completion as outlined in Ruff (1994).

2736-ZA Building

Discussion. The review effort included the seismic evaluation of the 2736-ZA structure as documented in Giller (1994b). The 2736-ZA building houses the plutonium storage vaults ventilation system providing two distinct functions: confinement and decay heat removal. Westinghouse Hanford Company classified the structure as a Safety Class 3 structure. Westinghouse Hanford Company used an equivalent static method to evaluate the seismic load on the facility. The analysis resulted in positive margins of safety with respect to material design strengths for a 0.2 g zpa (SDC 4.1) earthquake and damping ratio of 10%. The 10% damping ratio and the associated F_u factors are allowed by UCRL-15910 since all reinforcing details meet applicable ACI and UBC requirements.

The seismic analysis of the 2736-Z ventilation exhaust system housed within the 2736-ZA structure is presented in Ocoma (1994). The ventilation exhaust system is comprised of the ventilation ducting connecting the 2736-Z building to the 2736-ZA building, the exhaust fan, two stages of HEPA filters, the exhaust stack, and the diesel generator which provides a secondary source of power to the exhaust fan. Westinghouse Hanford Company categorized the ventilation exhaust system as Safety Class 3 equipment. The analyses for the exhaust stack, exhaust fan support, and HEPA filter housing were performed using masses, accelerations, and support forces taken from certified vendor information which provided the dynamic analyses performed by the vendor, Mason Industries Inc. The analysis for the diesel generator used the load from certified vendor information to verify the adequacy of the diesel generator anchorage. Operability for both the diesel generator and the exhaust fan was verified using the SQUIG/GIP method. The analysis of the ventilation ducting connecting the 2736-Z Building to the ventilation exhaust system located in the 2736-ZA Building resulted in a negative safety margin. Westinghouse Hanford Company's structural assessment group concluded that "...seismic motion of the 2 buildings may cause some shear tearing in the light gage steel duct. Very small holes may open after the shear tearing. The duct will remain in place after an earthquake."

During the review a concern regarding the results of the seismic evaluation of the duct connecting the 2736-Z and 2736-ZA buildings was raised. Specifically, what would be the potential release resulting from the small release path that may occur during the tearing of the duct? To alleviate this concern WHC performed an analysis (refer to FSAR Section 9.2.4A.6.3.6.E) to demonstrate that such a release would be insignificant. Westinghouse Hanford Company's safety analysts identified only one potential seismically related release scenario within the storage vaults. The scenario assumes that an operator fails to properly latch one of the lard can storage rack retaining cables such that it swings open during a seismic event, thus allowing 6 cans to fall to the floor. Westinghouse Hanford Company demonstrated that the falling lard cans would not pose a criticality concern and a release could only occur if both the inner and outer cans were to fail. Westinghouse Hanford Company conservatively demonstrated that the respirable release of plutonium to the vault would be insignificant (0.006 grams) assuming the simultaneous failure of both inner and outer can. The respirable release fraction was based on the free fall of a powder from a height of 2 meters and was reviewed against data presented in Mishima (1993). The analysis was reviewed and found to be acceptable.

Evaluation Findings. The DOE review concludes that the seismic evaluation of the 2736-ZA Building and the 2736-Z exhaust ventilation system is consistent with guidelines provided in UCRL 15910, and is therefore acceptable. The tearing which may occurring in the duct connecting the 2736-Z and 2736-ZA buildings was found not to present any significant increase in risk or consequences during the seismic event.

232-Z Building Methodology

Discussion The DOE review included WHC's seismic assessment of the 232-Z building presented in Ryan (1994) and Appendix 9.C of the PFP FSAR. The facility has not been operated for about 20 years and is destined for decontamination and decommissioning. It has been determined that building 232-Z has a maximum of 5900 grams of residual plutonium. To demonstrate adequate safety based on radiological consequences, WHC performed calculations to demonstrate the building would not collapse at the 0.09 g zpa level (i.e. the zpa for a general use type facility at the Hanford site). It is assumed that the building collapses at higher zpa's. Because of inherent conservatism in the analysis methods used to qualify acceptable seismic performance (usually insignificant damage), a higher seismic hazard must be used in an analysis which demonstrates that the building will not collapse. Westinghouse Hanford Company performed an analysis using an equivalent static method and a zpa of 0.2 g.

The 232-Z building is 56ft 8 in long and 36 ft 8 in wide and has a sloping roof the top of which is 15 ft to 15 ft 8 in. above the floor. The walls are made of 8 in. of concrete block with vertical and horizontal steel. The ceiling is 4 inches thick reinforced concrete on corrugated steel decking. There are four concrete beams that run east to west at the building column lines. The concrete beams and roof were made in a single pour and, with the decking, form a large slab.

A glovebox incinerator is located in the first floor process area, and was used to incinerate dry waste from which radioactive materials were then recovered. Appendix 9.C of the FSAR describes the damage suffered by the glovebox due to toppling during a seismic event. A comparison with a drop test of a shipping container was conducted, and WHC concluded that the glovebox damage is less than that of the shipping container. This was done to determine the bounding credible release for a greater than 0.09 g zpa seismic event.

Evaluation Findings The DOE review concludes that the 232-Z building structure will not collapse when subjected to 0.09 g zpa and finds WHC supporting documentation acceptable. Based on the accident scenarios, the DOE also believes that a 0.09 g seismic event will not result in a significant damage to the glovebox, and there will be no significant release for this seismic hazard.

9.2.4A.2 Postulated Seismic-Induced Accident Scenarios

(see FSAR Section 9.2.4A.2 *Seismic Accident Scenario*)

Discussion. Westinghouse Hanford Company performed what may be considered a seismic hazards assessment for the facility. The following facilities were considered in the assessment: 234-5Z, 236-Z, 241-Z, 242-Z, 232-Z, 291-Z, 2736-Z, -ZB, and -ZA, and various outside tanks. The assessment describes the postulated room-by-room damage that may occur as a result of a Design Basis Earthquake (DBE). Only those postulated scenarios that could potentially cause the dispersal of plutonium were included. The potential for seismically

induced criticalities and fires was also documented in the FSAR. To determine a bounding (i.e. worst case) seismic scenario, WHC assumed that the following processes were operating: RMC line; PRF solvent extraction; PRF miscellaneous treatment; Product handling; and Vault storage. Other explicit assumptions made in the analysis are as follows:

- gloveboxes, hoods, and conveyors specifically excluded from the analysis because they are inactive are listed in Table 9.2.4-3 of the FSAR,
- there is no processing activities in rooms 153, 145, and 149 of 234-5Z,
- the Analytical Labs only contains nominally one pound propane bottles, and the facility is limited to no more than two bottles located in a single room and 6 bottles in the facility,
- plutonium inventories are controlled in accordance with LCO 3.3.1 and AC 5.22,
- and no production of plutonium metal is occurring.

The various seismically induced scenarios were reviewed. The review determined that sufficient detail and justification were provided to ascertain that the predicted level of damage was appropriate. Typically, if a glovebox, component, or system has sufficient documentation to prove its seismic qualification then there was no postulated release. If on the other hand there was insufficient evidence to credit its seismic qualification then some release was assumed.

Evaluation Findings. The DOE review concludes that WHC has adequately documented the potential accident scenarios that may occur at PFP as a result of a DBE. Westinghouse Hanford Company has identified those gloveboxes and fume hoods in the facility which are, and are not, seismically qualified and has also conservatively identified the maximum quantity of dispersible plutonium which may be present in each glovebox and/or hood (based on criticality prevention specifications limits).

9.2.4A.3 Frequency of Design Basis Earthquake (DBE)

(see FSAR Section 9.2.4A.5 *Probability*)

Discussion. A 0.2 g pga seismic event is specified as the design basis earthquake (DBE) for the Hanford Site in accordance with SDC 4.1 (WHC 1991b). The DBE is shown to have a 10,000 year return period per UCRL-53582 or an annual probability of exceedance equal to 10^{-4} /yr. Westinghouse Hanford Company has stated that a 0.2 g pga seismic event is conservative with respect to their risk acceptance guidelines; however, from a safety analysis standpoint non-seismically qualified gloveboxes, fume hoods, and process equipment may be assumed to fail when subjected to seismic loading less than 0.2 g (i.e., smaller earthquakes with a return period less than 10,000 years). In this case a release may occur at a greater annual probability than the seismic analysis results presented in the FSAR. The DOE review

considered such a situation and concluded that 1) sufficient conservatism exists in the analyses to account for such errors, and 2) because a large fraction of the total radiological dose can be attributed to the failure of a single structure (232-Z Building), which is currently scheduled for decontamination and decommissioning activities, the concern does not constitute a major issue needing resolution.

Evaluation Findings. Westinghouse Hanford Company's definition of a design basis earthquake (0.2 g pga seismic event with an annual probability of exceedance equal to $10^{-4}/\text{yr}$) is consistent with guidelines developed in UCRL-15910 and is therefore acceptable.

9.2.4A.4 Quantity of Material Released into Facility during a DBE (see FSAR Section 9.2.4A.6 *Seismic Accident Analysis*)

In Section 9.2.4A.6 of the PFP FSAR, WHC presented conservative analyses regarding the amount of respirable plutonium that may be released into the facility during a DBE. Also presented are discussions regarding seismically induced criticalities and fires, and toxic release scenarios.

9.2.4A.5 Accident Analysis Assumptions (see FSAR Section 9.2.4A.6.1)

Discussion. Releases were developed based on the seismic induced failures documented in FSAR Section 9.2.4A.2, *Seismic Accident Scenario*, and the facility's established inventory limits. Westinghouse Hanford Company developed release fractions for four specific cases. Each case was reviewed and the release fractions were determined to have good technical basis. The four specific cases were as follows:

Case 1 - postulated scenario involves a seismically qualified glovebox.

Case 2 - postulated scenario involves a glovebox not seismically qualified and is assumed to fall over.

Case 3 - postulated scenario involves a seismically qualified fume hood.

Case 4 - postulated scenario involves a fume hood that is not seismically qualified.

Evaluation Findings. The DOE review concludes that the release fractions used in the analyses are conservative and that they have sufficient technical basis, and are therefore acceptable. Furthermore, the DOE review concludes that WHC has provided a conservative calculation of the quantity of respirable plutonium released into the facility.

9.2.4A.6 Accident Scenarios-Fire

(see FSAR Section 9.2.4A.2.12)

Discussion. The basic premise of the seismic induced fire accident scenario is that the fire loading will be low such that there could only be small localized fires that would not suspend a significant amount of Pu (refer to Section 9.2.2, Fires). Based on the information presented, this assumption appears reasonable.

Westinghouse Hanford Company analysts assumed that there exists a 1 in 10 chance that a 0.2 g seismic event would result in several small fires occurring simultaneously. The quantity of plutonium consumed by the fires was based on a fraction (assumed to be 10%) of the quantity of plutonium consumed in the postulated RMC line fire. The result was a 0.2 g release of respirable plutonium to the inside of the building. Westinghouse Hanford Company conservatively applied a leak path factor of 1.0 to calculate the release out of the facility. Westinghouse Hanford Company conservatively assumed that the material consumed by the small fires to be Class Y material and applied a >10% Pu-240, 90th percentile unit inhalation dose conversion factor to calculate the radiological dose to the public and onsite worker. The 0.2 g release is added to the total seismic scenario release (see SER Section 9.2.4.12).

Evaluation Findings. Because of the difficulty in precisely determining the potential for seismically-induced fires, WHC analysts developed a bounding radiological dose consequence for a seismically-induced fire. While no technical basis is provided to support WHC's assumptions regarding the quantity of plutonium involved in a fire, nor the 1 in 10 chance of a seismically induced fire occurring, they appear to be reasonable. Furthermore, DOE believes that WHC has demonstrated sufficient conservatism in their safety analyses of the seismic event, and risk acceptance guidelines, to accommodate possible errors resulting from such assumptions.

9.2.4A.7 Seismic-Induced Criticality

(see FSAR Section 9.2.4A.6.3.2.A.a)

Discussion. In analyzing the seismic event the contractor makes several simplifying assumptions to aid in the analysis; the assumptions are based on the premise that non-seismically qualified systems will fail in such a way as to maximize the release. The plant is assumed to be in an abnormal condition after the seismic event. The abnormal condition may include the following: normal flow streams may or may not be present; active valves (not manual) may open or close; lines may sever; tanks may fail; glovebox panels may fall out or crack; and pumps may or may not start operating and continue operation during the event. The seismically qualified portions of the building will remain standing but other portions of the building may collapse.

Seismic accident analysis for each room in the PFP was performed. A criticality accident analysis was presented for each of the following locations: 234-5Z, 2736-Z, and 2736-ZB storage vaults; glovebox HC-11; glovebox HC-18BS; glovebox HA-23S; glovebox HC-4;

glovebox HC-227S; water flooding in 2736-Z and 2736-ZB; glovebox HC-21C; and 236-Z. During the seismic event, it is possible for a criticality event to occur in the west gallery gloveboxes in the 236-Z building. The consequences from this event are bounded by the generic criticality event analyzed in 9.2.3 of the FSAR and reviewed in 9.2.3 of the SER. The criticality analysis for each of the other locations indicate that no seismically induced criticalities are possible in these locations.

Evaluation Findings. The initial assumption that non-seismic equipment fails in the worst possible condition can be interpreted to produce a scenario that could result in a criticality in Gloveboxes HC-4 and HC-27. The criticality is only possible if several unlikely events in non-seismic qualified equipment were to occur that include: failure of the top glovebox panels, loss of integrity of the side and bottom glovebox panels, failure of a firewater line that pours water into the gloveboxes, and failure of the criticality drain from falling debris. No formal analysis were presented to demonstrate that the probability of a seismically induced criticality occurring in glovebox HC-4 and/or HC-27 is less than 1.0×10^{-6} . However, the initial assumptions provided in the analysis are overly conservative in this case and, therefore, the analysis is acceptable.

9.2.4A.8 Hydrogen Gases Produced in Plutonium-HNO₃ Solutions (see FSAR Section 9.2.4A.6.3.2.A.n)

Discussion. Westinghouse Hanford Company postulated a sequence of events that may occur following an earthquake resulting in the loss of ventilation and isolation of plutonium storage tanks such as TK-45, thus causing an accumulation of hydrogen gas in the 1/2" vent lines leading to the ventilation system. Specifically, the analysis assumes the gas in the vent line builds up to greater than 4 percent concentration and is ignited by a source in the ventilation system. The ignition of hydrogen gas pressurizes the storage tank causing plutonium nitrate solution to be forced from the system and onto the floor. The result is a release of 0.009 grams of respirable plutonium into the 236-Z Building.

Evaluation Findings. The analysis is conservative. A hydrogen gas concentration of greater than 4 percent is required for ignition of gas in the vent line. Viscous and inertia effects in the liquid lines were neglected in the analysis. The glove box at the point of discharge is assumed to be breached as a result of the earthquake. The estimate of respirable plutonium is conservative, probably by at least an order of magnitude, perhaps even greater. The analysis is appropriately conservative and thus acceptable.

9.2.4A.9 Loss of Decay Heat Removal (see FSAR Section 9.2.4A.6.3.6.E)

Discussion. The safety analysis identified as a possible safety concern the loss of 2736-Z ventilation during a seismic event. The ventilation system provides for the removal of decay heat from the plutonium storage vaults located in the 2736-Z building. Natural radioactive decay of plutonium stored in containers produces a heat load in the vaults that must be

removed, otherwise containers will heat up and at elevated temperatures their seals may rupture. Westinghouse Hanford Company performed a heat transfer analysis regarding container heatup due to vault ventilation failure. The analysis assumed a 48 h heatup time, 45 containers per cubicle (the maximum container loading), each container generating 15 Watts (the maximum heat generation rate based on packaging restrictions).

The heat transfer analysis resulted in a container surface temperature of approximately 320 deg. F and a container pressure of 22 psia (or 7 psig) 48 h after a loss of vault cooling. Hydrostatic test results on the containers show that the containers rupture at approximately 95 psig. Based on this evidence WHC has concluded that there exists sufficient time to regain vault cooling if it is lost during an earthquake, and therefore, no OSRs or ACs have been developed to mitigate or eliminate the loss of vault cooling.

Evaluation Findings. An independent heat transfer calculation verified WHC's results (McKinnon 1993). The DOE review concludes that a loss of vault cooling does not pose an immediate danger to the Pu storage containers and that PFP has sufficient time to regain cooling to the vaults in the event that cooling is lost during a seismic event. Furthermore, seismic analyses have been prepared by WHC to demonstrate operability of an emergency diesel generator located in the 2736-ZA building and designed to provide backup power to the seismically qualified ventilation exhaust fans.

9.2.4A.10 Release from the 291-Z downstream ducting (see FSAR Section 9.2.4A.6.3.9)

Discussion. The ventilation ducting down stream of the final HEPA filters leading to the 291-Z stack is contaminated with an estimated 10 g of plutonium. This estimate was based on surveys taken from surfaces in the filter room and internal ducting. Analyses of these samples indicate that the material consists primarily of 6% ^{240}Pu material and is tightly adhered to the surfaces. Westinghouse Hanford Company assumed that some release will occur from this section of ducting as a result of vibratory motion during a seismic event. Westinghouse Hanford Company estimates that one-tenth of one percent of the material (0.01 g) is released during the seismic event based on a derivation predicting the material released from the surface of ducting up stream of the HEPA filters (see FSAR Section 9.2.4A.6.3.1.W, *Accident Analysis-Calculations, Downstream Plutonium*). The derivation demonstrates that 0.00012 g of Pu are suspended in a duct as a result of vibratory motion affecting 20 g of Pu attached to the duct surfaces up stream of the HEPA filters. Westinghouse Hanford Company used the results of this derivation, allowed two orders of magnitude for conservatism, and concluded that the release of material down stream of the HEPA filters would not exceed 0.01 g.

Evaluation Findings. The DOE review concludes that WHC has adequately demonstrated that a release from the ducting down stream of the final stage HEPA filters will not exceed 0.01 g of 6% ^{240}Pu . The potential release of material from ducting down stream of the final

stage HEPA filters has been added to the total release for the "ventilation" case in the seismic scenario.

9.2.4A.11 Release from the 232-Z Building

(see FSAR Section 9.2.4A.6.3.10)

Discussion. The 232-Z Building was used from 1960 to 1973 to incinerate combustible materials contaminated with plutonium for eventual recovery of the plutonium from the resulting ash. The facility has not been operated during the last 20 years and is currently scheduled for decontamination and decommissioning (D&D). In preparation for D&D, nondestructive assay (NDA) measurements were performed in the late 1980s. The NDA results concluded that the maximum plutonium inventory within the facility was 848 grams. Recently, new NDA measurements were performed to assess the clean out progress. These more recent measurements revealed that an error had been made with respect to the earlier measurements. The discovery prompted WHC to declare a Unreviewed Safety Question (USQ) since the actual plutonium inventory appeared to be greater than that analyzed in the FSAR. A bounding value of 5,900 grams was established in September 1994.

In 1991 when WHC issued the first version of the PFP FSAR (1991), neither the 232-Z structure, nor the incinerator glovebox located within, had any supporting seismic analyses and therefore the structure and glovebox were simply assumed to collapse during the seismic event. A very simple model was developed to determine a bounding release for the building.

To resolve the USQ, WHC proceeded along several parallel paths. A more detailed safety analysis was prepared to determine the release from the facility and at the same time a seismic evaluation of the building was performed and plans were made to seismically upgrade the incinerator glovebox anchorages.

The DOE review focused on WHC's most recent analysis regarding the release of material from the 232-Z building presented in FSAR Chapter 9, Appendix 9.C. The maximum plutonium inventory was used in the analysis. The respirable release fractions were reviewed against data provided in Mishima (1993) and determined to be bounding. The leak path factors were reviewed and are considered reasonable based on both the scenario assumed and WHC's most recent seismic evaluation efforts. The atmospheric dispersion coefficient, based on a ground level point release Gaussian plume model, represents the 99.5% worst-case dispersion and is conservative. The safety analysis conservatively results in 0.88 grams of plutonium being released from the 232-Z building during a seismic event resulting in 7.10 rem EDE exposure to the onsite receptor and 0.14 rem EDE exposure to the offsite receptor. In Section 9.2.4.1, it is concluded that the 232-Z structure will not collapse when subjected to a 0.09 g zpa seismic event which has an annual event frequency of $2 \times 10^{-3}/\text{yr}$. Based on WHC's onsite and offsite risk acceptance guidelines the consequences associated with a release from the 232-Z building are shown to be acceptable for an event frequency of $2 \times 10^{-3}/\text{yr}$. The 232-Z Building's contribution to the total release associated with a seismic event has been included and is presented in Section 9.2.4.6.

Evaluation Findings. The DOE review concludes that WHC has conservatively established the quantity of plutonium released from the 232-Z Building during a seismic event and has shown that the radiological consequences associated with such an event are acceptable based on their risk acceptance guidelines. Furthermore, the DOE review concludes that the release from the 232-Z Building contributes to approximately 40% of the total release resulting from the DBE event. This building is currently scheduled for decontamination and decommissioning (D&D) activities and, therefore, the risk associated with this building is temporary and will be reduced as the building is decontaminated.

9.2.4A.12 Quantity of Material Released to the Environment during a DBE (see FSAR Section 9.2.4A.7 *Release out of Facility*)

Westinghouse Hanford Company developed two separate release scenarios for the seismic event. The first assumes that an earthquake occurs and disrupts the electrical power to the facility's main ventilation supply and exhaust fans and that the backup steam supplied fans also fail to operate. The release in Case 1 is assumed to be caused by wind blowing through open doors, building cracks, etc. and is a fraction of the quantity of plutonium released into the facility as identified in FSAR Section 9.2.4A.6, *Seismic Accident Analysis*. The second case assumes that the earthquake does not disrupt the electrical power supplied to the fans, but the supply ventilation fans are shutdown by an installed seismically qualified seismic shutdown switch. The plutonium released in Case 2 is transported out of the facility through the 291-Z stack and the quantity released is again a fraction of the plutonium released into the facility as identified in FSAR Section 9.2.4.12, *Seismic Accident Analysis*.

9.2.4A.12a Case 1 - Both Supply and Exhaust Ventilation System Shutdown (see FSAR Section 9.2.4A.7.1 *"No Ventilation" Case*)

Discussion. Case 1 assumes that neither the supply nor exhaust ventilation system is operating after a seismic event. The analysis relies heavily on the seismic shutdown switch to prevent operation of the supply fans in the event of an earthquake. Operation of the supply fans without the exhaust fans running would provide a motive force to drive the dispersed plutonium out of the facility. The exterior doors located in the 234-5Z and 236-Z buildings were evaluated by WHC. Their construction was reviewed along with the seismic qualification of the door frame. The exterior doors located in the 234-5Z building are all assumed to open during a seismic event. Doors located in the 236-Z building are assumed to remain fully closed based on seismic qualification of the structure, therefore, there is no contribution from the PRF in Case 1. To estimate the release out of the facility a model was developed by WHC based on air flowing through the openings on one side of the building and out the other side. The analysis concludes that 0.68g of <10% ²⁴⁰Pu and 0.37g of >10% ²⁴⁰Pu are released to the atmosphere from various rooms within the 234-5Z building.

Evaluation Findings. The DOE review concludes that the analysis of the quantity of material released to the atmosphere during a seismic event for the "no ventilation" case is acceptable. The model is developed in sufficient detail with a sufficient level of conservatism

to adequately bound the release of respirable material out of the facility. The seismic ventilation shutdown switch identified by WHC is a Safety Class 1 SSC and is required to ensure the supply system is shutdown to prevent the possibility of pressurizing the building (assuming the exhaust failed) resulting in a greater quantity of plutonium being dispersed to the environment. The seismic shutdown switch was reviewed and found to consist of two redundant channels of instrumentation, each channel capable of securing the supply ventilation system in event of an earthquake. The seismic shutdown switch, a Safety Class 1 SSC, is seismically qualified and meets single failure criteria.

9.2.4A.12b Case 2 - Supply Ventilation System Shutdown, Exhaust Running (see FSAR Section 9.2.4A.7.2 "*Ventilation*" Case)

Discussion. Case 2 assumes that the exhaust ventilation system continues to operate after a seismic event but the supply system is shut down by the installed seismic shutdown switch as designed. The release to the atmosphere is via the 291-Z stack and potentially involves all of the plutonium released into the both the 234-5Z and 236-Z buildings. The analysis conservatively assumes that the exhaust duct is intact but the HEPA filters have been dislodged from the frames so that no credit is taken for HEPA filtration. Also no credit is taken for particle deposition. These are both considered very conservative assumptions. The analysis concludes that 2.48g of <10% ^{240}Pu and 9.69g of >10% ^{240}Pu are released through the 291-Z stack into the atmosphere.

Evaluation Findings. The DOE review concludes that the analysis of the quantity of material released from the 291-Z stack to the atmosphere during a seismic event for the "ventilation" case is acceptable. The analysis is developed in sufficient detail with a reasonable level of conservatism to adequately estimate the release of respirable material out of the facility.

9.2.4A.13 Radiological Dose Consequences for the DBE (see FSAR Section 9.2.4.1 *Dose Calculations*)

Discussion. Radiological dose calculations are presented for both of the cases analyzed. The consequences associated with Case 1 are calculated assuming a ground level point release (refer to SER Section 9.0.4.1) and average unit inhalation dose conversion factors for both <10% and >10% ^{240}Pu material (refer to SER Section 9.0.4.2). Atmospheric dispersion coefficients were based on nine year averaged Hanford site specific joint frequency meteorology data to calculate 99.5% worst case $\bar{\chi}/Q'$ values. The consequences associated with Case 2 are calculated assuming an elevated release from the 291-Z stack and also average unit inhalation dose conversion factors to describe the <10% and >10% ^{240}Pu material. The release from the 232-Z Building is also included in the total seismic dose (the 232-Z dose calculation is based on average unit inhalation dose conversion factor for <10% ^{240}Pu and a ground level point release). The resulting onsite and offsite doses are provided for both cases below and meet WHC's risk acceptance guidelines for an event frequency of $10^{-4}/\text{yr}$.

Case 1 "No Ventilation" - Supply and Exhaust Ventilation Systems Shutdown

Facility	Pu Release (g)	Onsite Dose (rem EDE)	Onsite RAGs @ 10 ⁻⁴ /yr (rem EDE)	Offsite Dose (rem EDE)	Offsite RAGs @ 10 ⁻⁴ /yr (rem EDE)
PFP	1.06	8.10	--	0.17	--
232-Z	0.88	7.10	--	0.14	--
Total	1.94	15.20	23	0.31	2.8

Case 2 "Ventilation" - Supply Ventilation System Shutdown, Exhaust Running

Facility	Pu Release (g)	Onsite Dose (rem EDE)	Onsite RAGs @ 10 ⁻⁴ /yr (rem EDE)	Offsite Dose (rem EDE)	Offsite RAGs @ 10 ⁻⁴ /yr (rem EDE)
PFP	12.18	5.0	--	1.77	--
232-Z	0.88	7.10	--	0.14	--
Total	13.06	12.10	23	1.91	2.8

Assuming the most limiting of the two scenarios (i.e., the no ventilation case) and considering the possibility of both a seismic-induced criticality and fire occurring concurrently would result in larger radiological dose consequences. This issue was raised during the review process. To account for this situation WHC simply totaled the radiological dose consequences and assumed that such an event would occur with a frequency of 10⁻⁶/yr to determine the acceptability of risk. The results are provided below and show that the combined risks are acceptable for a 10⁻⁶/yr event. The "no-ventilation" case results in the most limiting onsite dose, however, it is the "ventilation" case that results in the maximum offsite radiological dose consequences. The maximum offsite dose resulting from the "ventilation" case for the 10⁻⁴ event is 1.91 rem EDE and for the 10⁻⁶ event (including a concurrent seismically-induced fire and criticality) is 2.07 rem EDE.

The underlying assumption made by WHC is that there exists a 1 in 10 chance that a 0.2 g seismic event will cause a fire in the facility and that the release from such a fire would be bounded by the results presented in SER Section 9.2.4A.6. Likewise, WHC has assumed that there exists a 1 in 10 chance that a 0.2 g seismic event will cause a criticality to occur. While WHC's assumptions have been presented qualitatively and are not supported in a quantitative manner they appear to be reasonable. Furthermore, DOE believes that WHC has demonstrated sufficient conservatism in their safety analyses and risk acceptance guidelines to allow for possible errors resulting from such assumptions. For example, the combined dose

consequences remain below WHC's risk acceptance guidelines for the combined seismic event, fire, and criticality even at a frequency of 10^{-4} /yr.

No ventilation case with concurrent seismically-induced criticality and fire

Scenario	Pu Release (g)	Onsite Dose (rem EDE)	Onsite RAGs @ 10^{-6} /yr (rem EDE)	Offsite Dose (rem EDE)	Offsite RAGs @ 10^{-6} /yr (rem EDE)
PFP and 232-Z	1.94	15.20	--	0.31	--
Fire	0.2	5.32	--	0.11	--
Criticality	--	0.73		0.05	
Total	--	21.25	100	0.47	25

Evaluation Findings. The DOE review concludes that the radiological dose calculations were correctly performed based on the quantities of plutonium released as identified in FSAR Section 9.2.4A.7 *Release out of Facility*. The DOE review also concludes that the radiological dose consequence calculations have sufficient technical basis and provide a conservative bounding result. The risks associated with a 0.2 g zpa seismic event are acceptable based on the risk acceptance guidelines discussed in Section 9.0.2.2.

The following LCOs and ACs represent the minimum OSRs necessary to ensure that the risks associated with a DBE are maintained within the analyzed safety envelope:

LCO 3.1.1, *Criticality Prevention System*, requires the operability of specific criticality safety features at the facility.

LCO 3.2.3, *Supply Ventilation System Seismic Shutdown*, requires the operability of the supply ventilation seismic shutdown system to prevent pressurization of the 234-5Z building after a seismic event.

LCO 3.3.1, *Control of DISPERSIBLE Plutonium*, and AC 5.22, *Plutonium Inventory*, establishes inventory controls to ensure that facility inventory is equal to or less than the inventory assumed in the analysis.

AC 5.20, *Fire Protection*, establishes the requirements for implementing a fire protection program and identifies the key elements associated with such a program.

AC 5.21, *Flammable Gas Inventory*, establishes a program to control the storage and use of flammable gases within the facility.

Those SSCs listed in Table 4.3 that specifically provide preventive and/or mitigative features for the Seismic Event analysis are Building Structural Features, Ventilation System Seismic Shutdown System, and Plutonium Storage Arrays.

9.2.4A.14 Failure of the HNO₃ Storage Tank (see FSAR Section 9.2.4A.6.3.5.D)

The postulated event consists of a seismic event of sufficiently large magnitude to cause breach of the nitric acid tank in the 2735-Z area. The tank is assumed to contain the maximum capacity of 7000 gallons of nitric acid and the entire contents of the tank are released into the retention basin.

Discussion. The scope of the review is to highlight key assumptions, modeling approaches, limitations or uncertainties in the modeling approach, and consequence impact results for the HNO₃ storage tank failure analysis.

The maximum 7000 gallons of 57% solution of HNO₃ is instantaneously released and contained within the retention basin, forming a pool covering an area of 976 square feet. Temperature of the surrounding air and the liquid nitric acid spill are assumed to be 40°C. Ambient wind speed is taken to be 10 miles per hour and the Pasquil stability class is assumed to be class F.

The liquid pool evaporation model neglects the effects of solar radiation, ground conduction, and evaporative cooling. Neglecting the effects of evaporative cooling is a conservative approach. Evaporative cooling effectively reduces the temperature of the pool and thereby reduces the vapor pressure at the pool surface, which in turn results in smaller quantities of vapor entrained into the air passing above the surface of the pool. Using the properties of water in place of that of nitric acid (generally a conservative approach since water is more volatile than HNO₃), the effects of solar radiation and ground conduction (assuming that the ground temperature is 20°C greater than that of the liquid) are stated to have a net effect of raising the pool temperature approximately 2°C per hour. It is reasonable to conclude that the marginal increase in pool temperature would be compensated for by the effects of evaporative cooling.

The mass evaporation rate of the liquid nitric acid pool is estimated by means of an empirical correlation described by Clewell (1983). This correlation was derived from evaporation data obtained for hydrazine and employs a volatility factor in order to adjust the emission rate predicted by the correlation for chemicals other than hydrazine. The expression is given by,

$$Q = 0.08 u^{3/4} A Z [1 + (4.3 \times 10^{-3} T_p^2)]$$

where,

Q	=	evaporation rate (kg/hr);
u	=	wind speed (m/s);
A	=	spill area (m ²);
Z	=	volatility factor (dimensionless);
T_p	=	temperature of liquid pool (C).

Z is furthermore expressed as,

$$Z = \frac{P_{vx}M_x}{P_{vh}M_h}$$

with P_{vh} , P_{vx} denoting the vapor pressure of chemical species hydrazine (h) and the given species of concern (x), and M_h , M_x represent the molecular weights of hydrazine and the given species, x , respectively.

It was established via informal written responses by WHC that the evaporation rate predicted by the empirical correlation is conservatively greater than that of a mass transfer calculation where the effects of solar radiation, ground conduction and evaporative cooling have been neglected. Note that HNO_3 has a low vapor pressure at ambient temperatures. Chemical releases which possess a high volatility, such as cryogenic liquids, generally exhibit more complicated thermodynamic behavior, and hence the burden is placed on the analyst to ensure that a relatively simple correlation such as described above is indeed conservative in such cases.

The fraction of HNO_3 released to the atmosphere in vapor phase is simply determined by multiplying the constant evaporation rate obtained in the source term calculation by the total exposure time of interest. Since the steady-state, neutrally buoyant Gaussian plume model was used to estimate the downwind dispersion of nitric acid vapors, the exposure concentration levels are independent of time and therefore it is implicitly assumed that all the nitric acid will eventually evaporate.

The source Richardson, Ri_o , number was calculated for this specific release instance and compared to the following criteria:

$$\begin{aligned} Ri_o < 0.003 &\Rightarrow \text{passive dispersion} \\ Ri_o \geq 0.003 &\Rightarrow \text{negative buoyancy-dominated dispersion} \end{aligned}$$

The Richardson number as applied in the PFP FSAR is an approximation to the ratio of the rate at which stability suppresses the generation of turbulence to the rate at which turbulence is generated by mechanical shear forces. Using the source initial conditions at ambient temperature, the Richardson number was determined to be 2.3×10^{-6} , and therefore the passive Gaussian plume model was applied to the downwind dispersion analysis. A contaminant mass release rate of 8.6 g/s, wind speed of 10 miles per hour and stability class F complete the list of inputs necessary for estimation of downwind vapor concentration exposure levels.

The nearest onsite individual is exposed to a nitric acid vapor concentration of 1.8 ppm and the nearest offsite individual experiences a concentration of 0.032 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for HNO₃ at a frequency of one event per year (refer to Table 9.2 below), it follows that the risk due to a seismically induced HNO₃ spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies.

Table 9.2. Comparison of Risk Acceptance Guidelines to HNO₃ Tank Failure Concentration Estimates.

Annual Frequency (# events/yr)	Basis	HNO ₃ Concentration Risk Acceptance Threshold (ppm)	Estimated HNO ₃ Concentration (ppm)
Onsite			
1	ERPG-2	15	1.8
Offsite			
1	ERPG-1	2	0.032

Due to the relatively low estimates for nitric acid vapor concentration obtained for the nearest onsite and offsite individuals, the analysis can conservatively assume that the accident frequency is one per year and still satisfy the Toxicological Risk Acceptance Guidelines.

Evaluation Results. The nearest onsite individual is exposed to a nitric acid vapor concentration of 1.8 ppm and the nearest offsite individual experiences a concentration of 0.032 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for HNO₃ at a frequency of one event per year (refer Table 9.2), it follows that the risk due to a seismically induced HNO₃ spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies.

9.2.4A.15 Seismically-Induced Spill of CCL₄ (see FSAR Section 9.2.4A.6.3.5.E)

A total of 48 55-gallon drums of carbon tetrachloride are stored under a tent located between the 241-Z area and the PFP fence. The standard strong wind missile for hazard class 2 facilities assumed to be travelling at 50 miles per hour has been demonstrated to have a maximum damage capacity of penetrating 5 drums (Appendix A of the Central Waste Facility Safety Analysis Report, WHC 1989). Therefore collapse of the tent scaffolding above the

drums due to a seismic event would be less energetic than a high wind missile and hence 5 55-gallon drums are assumed to rupture and spill their contents into the sumps below the pallets on which the drums are resting.

Discussion. The scope of the review is to highlight key assumptions, modeling approaches, limitations or uncertainties in the modeling approach, and consequence impact results for the HNO₃ storage tank failure analysis.

The entire contents of 5 55-gallon drums or 275 gallons of CCl₄ are assumed to be released from confinement. The pallet sumps are assumed to capture enough liquid to fill 2/3 of their maximum capacity of 85 gallons. The remainder of the liquid is assumed to form a liquid pool on the ground surface 1 inch in depth. Temperature of the surrounding air and the liquid carbon tetrachloride are assumed to be 40°C. Ambient wind speed is taken to be 10 miles per hour and the Pasquil stability class is assumed to be class F.

The liquid pool evaporation model neglects the effects of solar radiation, ground conduction, and evaporative cooling. Neglecting the effects of evaporative cooling is a conservative approach. Evaporative cooling effectively reduces the temperature of the pool and thereby reduces the vapor pressure at the pool surface, which in turn results in smaller quantities of vapor entrained into the air passing above the surface of the pool. Ground conduction and solar radiation heat sources are generally of major importance when the boiling point of the liquid is below ambient temperature. Therefore, the high mass flow rate resulting in the evaporative cooling of the pool dominate the heat transfer effects. It is reasonable to conclude that the marginal increase in pool temperature due to ground conduction and radiative heat transfer would be compensated for by the effects of evaporative cooling.

Although CCl₄ has a relatively high volatility at the stated ambient temperature, its boiling point of 76.7°C is well above that of the 40°C assumed for the air temperature. Hence no flash atomization of aerosol particles are expected. Furthermore, the collapse of the scaffolding is not expected to introduce an impact event of sufficient energy to result in aerosol formation from mechanical shear.

The mass evaporation rate of the liquid nitric acid pool is estimated by means of an empirical correlation described by Clewell (1983). This correlation was derived from evaporation data obtained for hydrazine and employs a volatility factor in order to adjust the emission rate predicted by the correlation for chemicals other than hydrazine. The expression is given by

$$Q = 0.08 u^{3/4} A Z [1 + (4.3 \times 10^{-3} T_p^2)]$$

where

Q	=	evaporation rate (kg/hr);
u	=	wind speed (m/s);
A	=	spill area (m ²);
Z	=	volatility factor (dimensionless);
T_p	=	temperature of liquid pool (C).

Z is furthermore expressed as

$$Z = \frac{P_{vx}M_x}{P_{vh}M_h}$$

with P_{vh} , P_{vx} denoting the vapor pressure of chemical species hydrazine (h) and the given species of concern (x), and M_h , M_x represent the molecular weights of hydrazine and the given species, x , respectively.

It was established via informal written responses by WHC that the evaporation rate of CCl_4 predicted by the empirical correlation is conservatively greater than that of a mass transfer calculation where the effects of solar radiation, ground conduction and evaporative cooling have been neglected. Chemical releases which possess a high volatility, such as cryogenic liquids, generally exhibit complicated thermodynamic behavior, and hence the burden is placed on the analyst to ensure that a relatively simple correlation such as described above is indeed conservative in such cases.

The fraction of CCl_4 released to the atmosphere in vapor phase is simply determined by multiplying the constant evaporation rate obtained in the source term calculation by the total exposure time of interest. Since the steady-state, neutrally buoyant Gaussian plume model was used to estimate the downwind dispersion of CCl_4 vapors, the exposure concentration levels are independent of time and therefore it is implicitly assumed that all the carbon tetrachloride will eventually evaporate.

The source Richardson, Ri_o , number was calculated for this specific release instance and compared to the following criteria:

$$\begin{aligned} Ri_o < 0.003 &\Rightarrow \text{passive dispersion} \\ Ri_o \geq 0.003 &\Rightarrow \text{negative buoyancy-dominated dispersion} \end{aligned}$$

The Richardson number as applied in the PFP SAR is an approximation to the ratio of the rate at which stability suppresses the generation of turbulence to the rate at which turbulence is generated by mechanical shear forces. Using the source initial conditions at ambient temperature, the Richardson number was determined to be 4×10^{-3} which is marginally greater than 3×10^{-3} . Under these conditions it is reasonable to expect that the cloud will transition to neutrally buoyancy close to the release point. Furthermore, the 0.003 threshold limit distinguishing passive dispersion from negative buoyancy-dominated dispersion is judged to be conservatively low based on an analysis of empirical data presented by Britter (1980) and

discussed in further detail by Havens and Spicer (1985). Therefore, application of the passive Gaussian plume model is an appropriate approach for estimating downwind vapor concentrations. A contaminant mass release rate of 364 g/s, wind speed of 10 miles per hour and stability class F complete the list of inputs necessary for estimation of downwind vapor concentration exposure levels.

The nearest onsite individual is exposed to a carbon tetrachloride vapor concentration of 53 ppm and the nearest offsite individual experiences a concentration of 0.98 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for CCl₄ (refer to Table 9.3 below), it follows that the risk due to a seismically induced CCl₄ spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year for onsite individuals.

Table 9.3. Comparisons of Risk Acceptance Guidelines to CCl₄ Spill Concentration Estimates.

Annual Frequency (# events/yr)	Basis	CCl ₄ Concentration Risk Acceptance Threshold (ppm)	Estimated CCl ₄ Concentration (ppm)
Onsite			
1	ERPG-2	50	53
10 ⁻²	<i>Extrapolated</i>	100	53
10 ⁻⁴	<i>Extrapolated</i>	232	53
10 ⁻⁶	ERPG-3	500	53
Offsite			
1	ERPG-1	10	0.98
10 ⁻²	<i>Extrapolated</i>	17	0.98
10 ⁻⁴	<i>Extrapolated</i>	29	0.98
10 ⁻⁶	ERPG-2	50	0.98

The expected annual frequency of a seismic event of sufficient magnitude to cause collapse of the tent scaffolding and result in rupture of 55-gallon drums is approximately 10⁻⁴. Referring to Table 9.3, the estimated concentration of CCl₄ is well below the threshold concentrations for both onsite and offsite individuals.

Evaluation Results. The nearest onsite individual is exposed to a carbon tetrachloride vapor concentration of 53 ppm and the nearest offsite individual experiences a concentration of 0.98 ppm. Comparing both values to the Toxicological Risk Acceptance Guidelines for CCl_4 (refer to Table 9.3), it follows that the risk due to a seismically induced HNO_3 spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year for onsite individuals. The expected annual frequency of a seismic event of sufficient magnitude to cause collapse of the tent scaffolding and result in rupture of 55-gallon drums is approximately 10^{-4} . Referring to Table 9.3, the estimated concentration of CCl_4 is well below the threshold concentrations for both onsite and offsite individuals. Therefore the onsite and offsite consequences of a seismically induced CCl_4 spill meet the risk acceptance criteria.

9.2.4B Seismically Induced Deflagration in Glovebox MT-5

The original MT-5 safety analysis presented in the PFP FSAR (Draft 1991) was found to be unacceptable. The MT-5 accident scenario was identified as one of the nine open issues. The initial safety analysis concluded that a seismically qualified MT-5 hot-plate power cut-off switch should be installed to ensure that the power to the MT-5 dissolvers would be secured in the event of an earthquake. The switch was never installed and therefore the safety analysis was considered invalid. As a result, WHC released a draft of WHC-SD-CP-SAR-012 Rev 2, *MT-5 Safety Assessment* (October 1993). This safety assessment included a more detailed and rigorous analysis of the chemical processes occurring in the MT-5 dissolver pots. The analysis incorporated new laboratory studies which provided experimentally derived reaction rates relating to the chemistry in the MT-5 dissolver pots. The new analysis demonstrated that a seismically induced deflagration in MT-5 would result in low consequences, therefore, installation of a seismic shutdown switch would not be required. After several internal reviews by WHC's SEAC, and informal DOE reviews, a final MT-5 safety assessment document was issued in June 1994. The DOE review concluded that the safety assessment presented in WHC-SD-CP-SAR-012 Rev 2 was appropriate and acceptable. The results have been incorporated into the PFP FSAR (1994).

Discussion. The MT-5 glovebox is used primarily to convert plutonium-bearing residues to nitrate solution by dissolution in near-boiling nitric acid (12M HNO_3) and hydrofluoric acid (0.35M HF). The organic tributyl phosphate (TBP) is used in the solvent extraction process at the Plutonium Reclamation Facility (PRF) and is typically present in plant recycle feed processed in MT-5. At elevated temperatures (from 130 to 150 °C) and in the presence of HNO_3 or heavy-metal nitrates, TBP can exothermally decompose resulting in the rapid evolution of combustible gases, primarily butene. If the generation of flammable gases is significant, an ignition source may cause a deflagration that could potentially breach the MT-5 glovebox thus discharging plutonium-bearing materials. The hot plate control system provides several safety features to prevent such an event during normal operations.

Although the deflagration of flammable gases in MT-5 during normal operations was determined to be an incredible accident scenario (refer to SER Section 9.2.1C) a seismic event

has the potential to cause a common mode failure of the hot plate controller systems. For this reason WHC performed a seismically induced deflagration analysis.

Westinghouse Hanford Company presents a likely scenario in which an earthquake occurs and the operator flees without shutting down the MT-5 processes. It is conservatively assumed that the earthquake induces a combination of failures in the controller circuitry that results in a maximum current continuously flowing to the hot plate. The solution then heats up in an uncontrolled manner, begins to boil, and produces a significantly greater quantity of butene gas. In the 1991 draft version of the PFP FSAR, this analysis concluded that a seismically qualified shutdown switch should be installed to prevent such an event. The shutdown switch was never installed and the safety analysis was therefore invalid.

The re-analysis of the seismically induced scenario provides an estimate of the likelihood of such an event occurring and a deterministic radiological dose consequence analysis. Westinghouse Hanford Company determined that the probability of a seismically induced deflagration of TBP decomposition products in MT-5 is incredible, $1.9 \times 10^{-7}/\text{yr}$ for the DBE. However, WHC has performed a deterministic analysis to determine the dose consequence of this accident. The deterministic analysis calculates the rate of butene gas generation based on the most recent laboratory data generated for this accident scenario. It also assumes that ventilation in the glovebox is not functioning and then uses the FATHOMS code to calculate the distribution of flammable gas concentration in MT-5. The analysis goes one step further assuming an ignition source ignites the flammable gases and calculates the resultant pressure caused by the ignited gases in the glovebox. Although the resultant pressure may not be great enough to cause a glove to rupture, WHC assumes that gloves attached to the glovebox fail and that radioactive material is released. The radiological consequences were found to be bounded by those resulting from a fire in the MT-5 glovebox (refer to SER Section 9.2.2C).

Evaluation Findings. DOE reviewers reviewed the most recent experimental data produced by WHC regarding the gas generation rates associated with reactions between TBP and nitric acid with uranium and thorium nitrates at elevated temperatures. Westinghouse Hanford Company worked with the DOE TOMSK II Lessons Learned Team to gain a better understanding of their chemistry results and other information relating to potentially violent exothermic reactions between TBP and nitric acid. Their work concluded that the amount of heat generated by the oxidation of TBP and concentrated nitric acid with thorium nitrate (a stand-in for plutonium nitrate), at temperatures similar to those expected in MT-5 dissolver pots, is not enough heat to cause a violent explosion in the open system. These reaction rates show that the quantity of gas generated in the MT-5 dissolver pots under worst-case circumstances is not great enough to cause a large explosion; however, a minor deflagration of flammable gases may occur.

The DOE review concludes that the MT-5 seismically induced deflagration analysis is appropriate and acceptable. Furthermore, the review concludes that the probability of a seismically induced deflagration in MT-5 is incredible and that the radiological consequences

associated with such an event, if it were to occur, are bounded by the MT-5 fire scenario (shown to be acceptable in SER Section 9.2.2C).

The following LCOs and ACs represent the minimum OSRs necessary to ensure that glovebox MT-5 operations are conducted within the analyzed safety envelope:

LCO 3.3.1, *Control of DISPERSIBLE Plutonium*, and AC 5.22, *Plutonium Inventory*, establishes inventory controls to ensure that facility inventory is equal to or less than the inventory assumed in the analysis.

AC 5.20, *Fire Protection*, establishes the requirements for implementing a fire protection program and identifies the key elements associated with such a program.

Those SSCs listed in Table 4.3 that specifically provide preventive and/or mitigative features for the MT-5 Seismically-Induced Deflagration analysis are Building Structural Features.

9.2.5 Strong Wind

9.2.5.1 HNO₃ Storage Tank Rupture.

The postulated event consists of a steady 90 mile per hour wind which propels a 15 pound, 2 inch × 4 inch plank at 50 miles per hour. The missile breaches the one inch diameter transfer piping at the bottom of the HNO₃ storage tank in the 2735-Z area. The tank is assumed to contain the maximum capacity of 7000 gallons of nitric acid and the entire contents of the tank are released into the retention basin.

Discussion. The scope of the review is to highlight key assumptions, modeling approaches, limitations or uncertainties in the modeling approach, and consequence impact results for the HNO₃ storage tank failure analysis.

The maximum 7000 gallons of 57% solution of HNO₃ is released via a one inch diameter transfer line into the retention basin. Ten percent (10%) of the liquid effluent is assumed to immediately aerosolize primarily due to turbulent air entrainment. The remaining liquid forms a pool covering the entire basin with an area of 976 square feet. Temperature of the surrounding air and the liquid nitric acid spill are assumed to be 40°C. Ambient wind speed is taken to be 10 miles per hour and the Pasquill stability class is assumed to be class F.

The liquid pool evaporation model neglects the effects of solar radiation, ground conduction, and evaporative cooling. Neglecting the effects of evaporative cooling is a conservative approach. Evaporative cooling effectively reduces the temperature of the pool and thereby reduces the vapor pressure at the pool surface, which in turn results in smaller quantities of vapor entrained into the air passing above the surface of the pool. Using the properties of water in place of that of nitric acid (generally a conservative approach since water is more

volatile than HNO_3), the effects of solar radiation and ground conduction (assuming that the ground temperature is 20°C greater than that of the liquid) are stated to have a net effect of raising the pool temperature approximately 2°C per hour. It is reasonable to conclude that the marginal increase in pool temperature would be compensated for by the effects of evaporative cooling.

It is difficult to characterize the amount of aerosolization from a pool due to a 90 mph wind. The most applicable experiments that WHC is aware of are discussed in DOE-HDBK-0013-93, section 3.2.5.1. The referenced experiments were performed by Robyler and Owczarski and measured the aerodynamic entrainment of salt aerosols from sea water as a function of wind speed and distance from the lee shore. Unfortunately, the highest wind speed considered was 15 m/s (33.6 mph). It is worth noting, however, that for a pool the size in the HNO_3 spill (100 m^2), the release rate is on the order of $1 \times 10^{-7} \text{ kg/s}$ for a 15 m/s wind. Thus, using the dispersion wind speed of 19 m/s (which is not too far removed from the experimental data) and increasing the aerosolization rate by an order of magnitude to cover uncertainties due to extrapolation, we would expect to see a release rate on the order of $1 \times 10^{-6} \text{ kg/s}$. This would be $1 \times 10^{-6} / 4.6 \times 10^{-2} \times 100\% = 0.002\%$ of the evaporation rate. Based on the data presented in DOE-HDBK-0013-93, the aerosolization rate increases by approximately an order of magnitude when the wind speed is doubled. Thus, extrapolating the 15 m/s data, the release rate for a 40 m/s wind is on the order of 0.1% of the evaporation rate. In order to ensure the worst possible credible conditions, the analysis was revised to consider a 10% aerosolization rate as the HNO_3 is released from the tank.

Standard outflow equations were used to determine the time dependent mass flowrate of HNO_3 from the transfer line. The maximum outflow rate was determined to be 5.725 kg/s. Assuming that 10% of this outflow is aerosol, and of this quantity 57% is actually HNO_3 , the maximum HNO_3 evolution rate from aerosol entrainment becomes 0.326 kg/s. The steady-state evaporation rate of 0.046 kg/s obtained from the model of Clewell (1983) described below is then added to the above maximum aerosol entrainment rate to obtain the total peak vapor release rate of 0.372 kg/s.

The mass evaporation rate of the liquid nitric acid pool is estimated by means of an empirical correlation described by Clewell (1983). This correlation was derived from evaporation data obtained for hydrazine and employs a volatility factor in order to adjust the emission rate predicted by the correlation for chemicals other than hydrazine. The expression is given by

$$Q = 0.08 u^{3/4} A Z [1 + (4.3 \times 10^{-3} T_p^2)]$$

where,

Q = evaporation rate (kg/hr);
 u = wind speed (m/s);
 A = spill area (m²);
 Z = volatility factor (dimensionless);
 T_p = temperature of liquid pool (C).

Z is furthermore expressed as,

$$Z = \frac{P_{vx}M_x}{P_{vh}M_h}$$

with P_{vh} , P_{vx} denoting the vapor pressure of chemical species hydrazine (h) and the given species of concern (x), and M_h , M_x represent the molecular weights of hydrazine and the given species, x , respectively.

It was established via informal written responses by WHC that the evaporation rate predicted by the empirical correlation is conservatively greater than that of a mass transfer calculation where the effects of solar radiation, ground conduction and evaporative cooling have been neglected. Note that HNO₃ has a low vapor pressure at ambient temperatures. Chemical releases which possess a high volatility, such as cryogenic liquids, generally exhibit more complicated thermodynamic behavior, and hence the burden is placed on the analyst to ensure that a relatively simple correlation such as described above is indeed conservative in such cases.

The fraction of HNO₃ released to the atmosphere in vapor phase is simply determined by multiplying the constant evaporation rate obtained in the source term calculation by the total exposure time of interest. Since the steady-state, neutrally buoyant Gaussian plume model was used to estimate the downwind dispersion of nitric acid vapors, the exposure concentration levels are independent of time and therefore it is implicitly assumed that all the nitric acid will eventually evaporate.

The source Richardson, Ri_o , number was calculated for this specific release instance and compared to the following criteria:

$$\begin{aligned}
 Ri_o < 0.003 &\Rightarrow \text{passive dispersion} \\
 Ri_o \geq 0.003 &\Rightarrow \text{negative buoyancy-dominated dispersion}
 \end{aligned}$$

The Richardson number as applied in the PFP SAR is an approximation to the ratio of the rate at which stability suppresses the generation of turbulence to the rate at which turbulence is generated by mechanical shear forces. Using the source initial conditions at ambient temperature, the Richardson number was determined to be 2×10^{-5} , and therefore the passive Gaussian plume model was applied to the downwind dispersion analysis. A contaminant mass release rate of 0.372 g/s, wind speed of 10 miles per hour and stability class F complete the list of inputs necessary for estimation of downwind vapor concentration exposure levels.

The nearest onsite individual is exposed to a nitric acid vapor concentration of 18 ppm and the nearest offsite individual experiences a concentration of 0.29 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for HNO₃ at a frequency of one event per year (refer to Table 9.4 below), it follows that the risk due to a high wind induced HNO₃ spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year for onsite individuals.

Table 9.4. Comparison of Risk Acceptance Guidelines to HNO₃ Tank Transfer Line Failure Concentration Estimates.

Annual Frequency (# events/yr)	Basis	HNO ₃ Concentration Risk Acceptance Threshold (ppm)	Estimated HNO ₃ Concentration (ppm)
Onsite			
1	ERPG-2	15	18
10 ⁻²	<i>Extrapolated</i>	19	18
10 ⁻⁴	<i>Extrapolated</i>	24	18
10 ⁻⁶	ERPG-3	30	18
Offsite			
1	ERPG-1	2	0.29
10 ⁻²	<i>Extrapolated</i>	3.9	0.29
10 ⁻⁴	<i>Extrapolated</i>	7.7	0.29
10 ⁻⁶	ERPG-2	15	0.29

The expected annual frequency of a high wind event of 90 miles per hour at the Hanford site is between 10⁻⁵ and 10⁻⁶ per year (UCRL-15910). The analysis conservatively assumes that the 90 mile per hour wind is assigned an annual frequency of 10⁻². Referring to Table 9.4, the estimated concentration of HNO₃ is well below the threshold concentrations for both onsite and offsite individuals.

Evaluation Results. The nearest onsite individual is exposed to a nitric acid vapor concentration of 18 ppm and the nearest offsite individual experiences a concentration of 0.29 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for HNO₃ at a frequency of one event per year (refer to Table 9.4), it follows that the risk due to a high

wind induced HNO_3 spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year for onsite individuals. Since a strong wind event of this nature has a estimated maximum frequency of 10^{-2} per year, the onsite and offsite consequences of a high wind induced nitric acid spill meet the risk acceptance criteria.

9.2.5.2 CCl_4 Drum Rupture.

A total of 48 55-gallon drums of carbon tetrachloride are stored under a tent located between the 241-Z area and the PFP fence. The standard strong wind missile for hazard class 2 facilities assumed to be travelling at 50 miles per hour has been demonstrated to have a maximum damage capacity of penetrating 5 drums (Appendix A of the Central Waste Facility Safety Analysis Report, WHC 1989). Collapse of the tent scaffolding above the drums due to missiles generated by high winds would be less energetic than direct contact by a high wind missile. Therefore, five drums are assumed to be penetrated by missile contact and another five drums are assumed to rupture due to collapse of the overhead scaffolding.

Discussion. The scope of the review is to highlight key assumptions, modeling approaches, limitations or uncertainties in the modeling approach, and consequence impact results for the failure of the CCl_4 storage drums.

The entire contents of 10 55-gallon drums or 550 gallons of CCl_4 are assumed to be released from confinement. The pallet sumps are assumed to capture enough liquid to fill 2/3 of their maximum capacity of 85 gallons. The remainder of the liquid is assumed to form a liquid pool on the ground surface 1 inch in depth. The 10 drums are assumed to be distributed over three pallets, and therefore 380 gallons of CCl_4 are assumed to fall onto the ground surface and form an evaporating pool. Temperature of the surrounding air and the liquid carbon tetrachloride are assumed to be 40°C . Ambient wind speed is taken to be 10 miles per hour and the Pasquill stability class is assumed to be class F.

The liquid pool evaporation model neglects the effects of solar radiation, ground conduction, and evaporative cooling. Neglecting the effects of evaporative cooling is a conservative approach. Evaporative cooling effectively reduces the temperature of the pool and thereby reduces the vapor pressure at the pool surface, which in turn results in smaller quantities of vapor entrained into the air passing above the surface of the pool. Ground conduction and solar radiation heat sources are generally of major importance when the boiling point of the liquid is below ambient temperature. Therefore, the high mass flow rate resulting in the evaporative cooling of the pool dominate the heat transfer effects. It is reasonable to conclude that the marginal increase in pool temperature due to ground conduction and radiative heat transfer would be compensated for by the effects of evaporative cooling.

Although CCl_4 has a relatively high volatility at the stated ambient temperature, its boiling point of 76.7°C is well above that of the 40°C assumed for the air temperature. Hence no flash atomization of aerosol particles are expected. Furthermore, the collapse of the

scaffolding is not expected to introduce an impact event of sufficient energy to result in aerosol formation from mechanical shear.

The mass evaporation rate of the liquid nitric acid pool is estimated by means of an empirical correlation described by Clewell (1983). This correlation was derived from evaporation data obtained for hydrazine and employs a volatility factor in order to adjust the emission rate predicted by the correlation for chemicals other than hydrazine. The expression is given by

$$Q = 0.08 u^{3/4} A Z [1 + (4.3 \times 10^{-3} T_p^2)]$$

where,

Q = evaporation rate (kg/hr);
 u = wind speed (m/s);
 A = spill area (m²);
 Z = volatility factor (dimensionless);
 T_p = temperature of liquid pool (C).

Z is furthermore expressed as,

$$Z = \frac{P_{vx} M_x}{P_{vh} M_h}$$

with P_{vh} , P_{vx} denoting the vapor pressure of chemical species hydrazine (h) and the given species of concern (x), and M_h , M_x represent the molecular weights of hydrazine and the given species, x , respectively.

It was established via informal written responses by WHC that the evaporation rate of CCl₄ predicted by the empirical correlation is conservatively greater than that of a mass transfer calculation where the effects of solar radiation, ground conduction and evaporative cooling have been neglected. Chemical releases which possess a high volatility, such as cryogenic liquids, generally exhibit complicated thermodynamic behavior, and hence the burden is placed on the analyst to ensure that a relatively simple correlation such as described above is indeed conservative in such cases.

The fraction of CCl₄ released to the atmosphere in vapor phase is simply determined by multiplying the constant evaporation rate obtained in the source term calculation by the total exposure time of interest. Since the steady-state, neutrally buoyant Gaussian plume model was used to estimate the downwind dispersion of CCl₄ vapors, the exposure concentration levels are independent of time and therefore it is implicitly assumed that all the carbon tetrachloride will eventually evaporate.

The source Richardson, Ri_o , number was calculated for this specific release instance and compared to the following criteria:

$$Ri_o < 0.003 \Rightarrow \text{passive dispersion}$$

$$Ri_o \geq 0.003 \Rightarrow \text{negative buoyancy-dominated dispersion}$$

The Richardson number as applied in the PFP SAR is an approximation to the ratio of the rate at which stability suppresses the generation of turbulence to the rate at which turbulence is generated by mechanical shear forces. Using the source initial conditions at ambient temperature, the Richardson number was determined to be 3.4×10^{-4} , and therefore the passive Gaussian plume model was applied to the downwind dispersion analysis. A contaminant mass release rate of 3.29 kg/s, wind speed of 10 miles per hour and stability class F complete the list of inputs necessary for estimation of downwind vapor concentration exposure levels.

The nearest onsite individual is exposed to a carbon tetrachloride vapor concentration of 64 ppm and the nearest offsite individual experiences a concentration of 1.0 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for CCl_4 (refer to Table 9.5 below), it follows that the risk due to a seismically induced CCl_4 spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year for onsite individuals.

Table 9.5. Comparisons of Risk Acceptance Guidelines to CCl_4 Drum Rupture Concentration Estimates.

Annual Frequency (# events/yr)	Basis	CCl_4 Concentration Risk Acceptance Threshold (ppm)	Estimated CCl_4 Concentration (ppm)
Onsite			
1	ERPG-2	50	64
10^{-2}	<i>Extrapolated</i>	100	64
10^{-4}	<i>Extrapolated</i>	232	64
10^{-6}	ERPG-3	500	64
Offsite			
1	ERPG-1	10	1.0
10^{-2}	<i>Extrapolated</i>	17	1.0
10^{-4}	<i>Extrapolated</i>	29	1.0
10^{-6}	ERPG-2	50	1.0

The expected annual frequency of a high wind event of 90 miles per hour at the Hanford site is between 10^{-5} and 10^{-6} per year (UCRL-15910). The analysis conservatively assumes that the 90 mile per hour wind is assigned an annual frequency of 10^{-2} . Referring to Table 9.5, the estimated concentration of CCl_4 is well below the threshold concentrations for both onsite and offsite individuals.

Evaluation Results. The nearest onsite individual is exposed to a carbon tetrachloride vapor concentration of 64 ppm and the nearest offsite individual experiences a concentration of 1.0 ppm. Comparing both values to the Toxicological Risk Acceptance Guidelines for CCl_4 (refer to Table 9.5), it follows that the risk due to a high wind induced CCl_4 drum rupture satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year for onsite individuals. The expected annual frequency of a seismic event of sufficient magnitude to cause collapse of the tent scaffolding and result in rupture of 55-gallon drums is approximately 10^{-2} . Referring to Table 9.5, the estimated concentration of CCl_4 is well below the threshold concentrations for both onsite and offsite individuals. Therefore the onsite and offsite consequences of a seismically induced CCl_4 spill meet the risk acceptance criteria.

9.2.6 Hydrogen Fluoride-Induced Plutonium Release

Currently, there are no production plans or operational reasons to use Hydrogen Fluoride in the facility and, therefore, this accident scenario has been omitted from the PFP FSAR. Furthermore, WHC has stated that Hydrogen Fluoride gas is no longer stored within the facility. If the plant's operational mission changes such that it is necessary to use Hydrogen Fluoride in plant processes, WHC shall perform a safety assessment and submit it to DOE for review.

9.2.7 Aircraft Accident

Discussion. Westinghouse Hanford Company considered the possibility of an aircraft accident adversely affecting the facility's confinement structures. Their analysis included a review of the types of aircraft flown in the vicinity of the Hanford site, the various air traffic patterns, and the location of nearby airports. The analysis also includes an appraisal of the airports and airways in the Hanford vicinity, FAA restrictions for air space usage over the Hanford site, and typical air traffic patterns over Hanford and the PFP for the following types of aircraft: commercial air carriers; general aviation; military; and contract chemical applicators. Westinghouse Hanford Company applied the methodology presented in NUREG-0800, *Standard Review Plan*, to estimate the total aircraft hazard probability at the facility.

Evaluation Findings. The aircraft accident analysis considered the combined probability associated with various aircraft hazards at PFP. The analysis employs methodology established in NUREG-0800, Section 3.5.1.6, *Aircraft Hazards*, to estimate the total annual probability of an aircraft related accident. The analysis concluded that the probability of an aircraft impacting the PFP is 1.28×10^{-8} /year, an incredible event. The DOE review concludes

that the aircraft accident analysis is acceptable since it was performed in accordance with methodology established in NUREG-0800.

9.2.8 Toxic Chemical Release (CCl_4)

A total of 48 55-gallon drums of carbon tetrachloride are stored under a tent located between the 241-Z area and the PFP fence. The postulated release scenario involves relocation of the drums via forklift. It is assumed that the tines are improperly positioned as the forklift approaches the pallet and that the tines puncture all four drums located on a single pallet. Furthermore it is assumed that the four punctured drums are pushed off the pallet and cause four adjacent drums to fall and begin to leak as a result of impact with the ground. Finally, as the forklift operator is removing the forklift, the forklift contacts the overhead scaffolding which subsequently falls and causes the breach of five additional 55-gallon drums. Thus, the contents of thirteen 55-gallon drums (715 gallons) of CCl_4 are spilled onto the ground and form an evaporating liquid pool.

Discussion. The scope of the review is to highlight key assumptions, modeling approaches, limitations or uncertainties in the modeling approach, and consequence impact results for the failure of CCl_4 storage drums.

The entire contents of 13 55-gallon drums or 715 gallons of CCl_4 are assumed to be released from confinement. No credit was taken for the pallet sumps capturing a portion of the liquid spill. The spill is assumed to form a liquid pool on the ground surface 1 inch in depth. Temperature of the surrounding air and the liquid carbon tetrachloride are assumed to be 40°C . Ambient wind speed is taken to be 10 miles per hour and the Pasquil stability class is assumed to be class F.

The liquid pool evaporation model neglects the effects of solar radiation, ground conduction, and evaporative cooling. Neglecting the effects of evaporative cooling is a conservative approach. Evaporative cooling effectively reduces the temperature of the pool and thereby reduces the vapor pressure at the pool surface, which in turn results in smaller quantities of vapor entrained into the air passing above the surface of the pool. Ground conduction and solar radiation heat sources are generally of major importance when the boiling point of the liquid is below ambient temperature. Therefore, the high mass flow rate resulting in the evaporative cooling of the pool dominate the heat transfer effects. It is reasonable to conclude that the marginal increase in pool temperature due to ground conduction and radiative heat transfer would be compensated for by the effects of evaporative cooling.

Although CCl_4 has a relatively high volatility at the stated ambient temperature, its boiling point of 76.7°C is well above that of the 40°C assumed for the air temperature. Hence no flash atomization of aerosol particles are expected. Furthermore, the collapse of the scaffolding is not expected to introduce an impact event of sufficient energy to result in aerosol formation from mechanical shear.

The mass evaporation rate of the liquid CCl_4 pool is estimated by means of an empirical correlation described by Clewell (1983). This correlation was derived from evaporation data obtained for hydrazine and employs a volatility factor in order to adjust the emission rate predicted by the correlation for chemicals other than hydrazine. The expression is given by

$$Q = 0.08 u^{3/4} A Z \left[1 + (4.3 \times 10^{-3} T_p^2) \right]$$

where,

Q	=	evaporation rate (kg/hr);
u	=	wind speed (m/s);
A	=	spill area (m^2);
Z	=	volatility factor (dimensionless);
T_p	=	temperature of liquid pool (C).

Z is furthermore expressed as,

$$Z = \frac{P_{vx} M_x}{P_{vh} M_h}$$

with P_{vh} , P_{vx} denoting the vapor pressure of chemical species hydrazine (h) and the given species of concern (x), and M_h , M_x represent the molecular weights of hydrazine and the given species, x , respectively.

It was established via informal written responses by WHC that the evaporation rate of CCl_4 predicted by the empirical correlation is conservatively greater than that of a mass transfer calculation where the effects of solar radiation, ground conduction and evaporative cooling have been neglected. Chemical releases which possess a high volatility, such as cryogenic liquids, generally exhibit complicated thermodynamic behavior, and hence the burden is placed on the analyst to ensure that a relatively simple correlation such as described above is indeed conservative in such cases.

The fraction of CCl_4 released to the atmosphere in vapor phase is simply determined by multiplying the constant evaporation rate obtained in the source term calculation by the total exposure time of interest. Since the steady-state, neutrally buoyant Gaussian plume model was used to estimate the downwind dispersion of CCl_4 , the exposure concentration levels are independent of time and therefore it is implicitly assumed that all the carbon tetrachloride will eventually evaporate.

The source Richardson, Ri_o , number was calculated for this specific release instance and compared to the following criteria:

$$\begin{aligned} Ri_o < 0.003 &\Rightarrow \text{passive dispersion} \\ Ri_o \geq 0.003 &\Rightarrow \text{negative buoyancy-dominated dispersion} \end{aligned}$$

The Richardson number as applied in the PFP SAR is an approximation to the ratio of the rate at which stability suppresses the generation of turbulence to the rate at which turbulence is generated by mechanical shear forces. Using the source initial conditions at ambient temperature, the Richardson number was determined to be 7×10^{-3} which is sufficiently greater than 3×10^{-3} such that dense gas effects may play a role in the dispersion of the vapors. A simple empirical model dense gas model attributed to Britter and McQuaid (1988) provides centerline concentrations relative to initial concentrations of the plume. The analysis recognizes that the model is only applicable to C or D stability wind fields, in contrast to the actual scenario hypothesis of a F wind stability class. In an effort to compensate for the lack of generality of the model, the analysis assumes that no dispersion occurs between the point of release and the downwind distance at which the Britter and McQuaid model predicts that the dense gas cloud will transition to a neutrally buoyant plume. The implied assertion here is that the degree of conservatism introduced by assuming no dilution of the dense gas compensates for the lack of applicability of the Britter/McQuaid model in the case of F stability wind conditions. The dense gas dispersion code available at WHC is DEGADIS, and it has not been properly certified for use in a SAR. Once the need for dense gas dispersion calculations was identified, insufficient time remained to obtain certification for any of the usual dense gas codes. As a confirming calculation, however, WHC has run the ALOHA emergency response code. ALOHA is based on DEGADIS and was benchmarked by the developers of NOAA against DEGADIS to ensure that only minor differences existed in results obtained from both models. The dense gas dispersion model in ALOHA was run with a direct source of 1.19 kg/s of CCl_4 as in Section 9.2.8 with the result that the onsite receptor located 550 m from the spill was subject to a maximum concentration of 48 ppm. This is a factor of three below the hand calculations presented in section 9.2.8.

Using the Britter and McQuaid model, a downwind distance of 125 meters was estimated over which the dense gas effects may cause the centerline concentration to be higher than what the standard Gaussian plume model would predict. Therefore, beyond 125 meters, the passive Gaussian plume model is applied, assuming no dispersion has taken place for distances of 125 m or less from the source point. A contaminant mass release rate of 1.19 kg/s, wind speed of 10 miles per hour and stability class F complete the list of inputs necessary for estimation of downwind vapor concentration exposure levels.

The nearest onsite individual is exposed to a carbon tetrachloride vapor concentration of 150 ppm and the nearest offsite individual experiences a concentration of 1.7 ppm. In general, the Toxicological Risk Acceptance Guidelines specify successively higher threshold limiting concentrations for respectively lower frequencies of event occurrence. Comparing both values to the Toxicological Risk Acceptance Guidelines for CCl_4 (refer to Table 9.6 below), it follows that the risk due to a handling accident induced CCl_4 spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year and 10^{-2} event per year for onsite individuals.

Table 9.6. Comparisons of Risk Acceptance Guidelines to CCl₄ Spill Concentration Estimates.

Annual Frequency (# events/yr)	Basis	CCl ₄ Concentration Risk Acceptance Threshold (ppm)	Estimated CCl ₄ Concentration (ppm)
Onsite			
1	ERPG-2	50	150
10 ⁻²	<i>Extrapolated</i>	100	150
10 ⁻⁴	<i>Extrapolated</i>	232	150
10 ⁻⁶	ERPG-3	500	150
Offsite			
1	ERPG-1	10	1.7
10 ⁻²	<i>Extrapolated</i>	17	1.7
10 ⁻⁴	<i>Extrapolated</i>	29	1.7
10 ⁻⁶	ERPG-2	50	1.7

The standard probability for an act of commission such as the forklift operator misaligning the tines is 10⁻³/demand. Because no additional CCl₄ is to be procured and current inventory of CCl₄ is not to be used, the forklift demand frequency was assumed to be one per year. Furthermore it is argued that an additional error on the part of the operator and/or equipment failure resulting in loss of forklift control would be necessary in order to affect collapse of the overhead scaffolding. The additional error was assigned a probability of 10⁻¹ per forklift, and hence the estimated annual probability of the above postulated accident taking place is 10⁻³/year × 10⁻¹ = 10⁻⁴/year. Referring to Table 9.6, the estimated concentration of CCl₄ is well below the threshold concentrations for both onsite and offsite individuals.

Evaluation Results. The nearest onsite individual is exposed to a carbon tetrachloride vapor concentration of 150 ppm and the nearest offsite individual experiences a concentration of 1.7 ppm. Comparing both values to the Toxicological Risk Acceptance Guidelines for CCl₄ (refer to Table 9.6), it follows that the risk due to a handling accident induced CCl₄ spill satisfies the Toxicological Risk Acceptance Guidelines for all event frequencies with the exception of an annual frequency of 1 event per year and 10⁻² event per year for onsite individuals. The expected annual frequency of a forklift accident which would result in the rupture of 13 55-gallon drums is approximately 10⁻⁴. Referring to Table 9.6, the estimated concentration of CCl₄ is well below the threshold concentrations for both onsite and offsite individuals. Therefore the onsite and offsite consequences of a handling accident induced CCl₄ spill meet the risk acceptance criteria.

9.3 UNCERTAINTIES ASSOCIATED WITH THE SAFETY ANALYSES

The uncertainties associated with the quantitative evaluation of radiological exposure were not explicitly discussed in the FSAR for the various accidents analyzed. Uncertainties related to the safety analyses can be divided into two groups; those uncertainties associated with the calculation of radiological dose consequences, and those associated with the probability of an event occurring.

Westinghouse Hanford Company did not determine the uncertainties associated with the exposure calculations since they assumed conservative source term values, airborne release fractions (ARF), respirable fractions (RF), and inhalation dose conversion factors. The review effort paid particular attention to the assumptions made by WHC in calculating the radiological and toxicological consequences associated with each accident. The reviewers conclude that bounding values were used to describe the source term, ARF, RF, and inhalation dose conversion factors and that it is appropriate not to present uncertainty values.

9.4 VERIFICATION AND VALIDATION OF COMPUTER CODES

The DOE review effort also ensured that computer codes used in performing the various analyses had associated with them a proper verification and validation program. A variety of computer codes were used in the accident analyses. A multi-phase thermal hydraulic code, FATHOMS⁴, was used in analyzing the MT-5 deflagration. A probabilistic risk assessment code, IRRAS, was used to determine the frequency of the most bounding fire scenario. An atmospheric dispersion code, GXQ, was used to determine the atmospheric dispersion coefficients to be applied in calculating doses. And a variety of structural analysis codes were used to determine the consequences associated with the design basis earthquake scenario.

FATHOMS⁴ is a multi-phase, multi-dimensional, thermal hydraulic computer code. The code has been accepted by DOE for use in level 1 documents. The Integrated Reliability and Risk Analysis System (IRRAS) code is a state-of-the-art, microcomputer based PRA model development and analysis tool used by the nuclear industry to address key nuclear plant safety issues. Westinghouse Hanford Company used this code to determine the probability of occurrence for both the MT-5 and laboratory waste concentrator explosions, and also the bounding fire scenario. The IRRAS program was developed for DOE by Idaho National Engineering Laboratory (INEL). After contacting the developers at INEL it was determined that version 2.0 of this code has never undergone a formal verification and validation process; however, the code is referenced in much of the literature and many recent PRA codes use IRRAS to bench mark their performance. Because the code is recognized in the industry, its use in the PFP FSAR is considered acceptable.

⁴ ©FATHOMS is copyrighted by Numerical Applications, Incorporated

The GXQ code is currently in use by Westinghouse Hanford Company's Safety Analysis and Regulation (SAR) department for performing safety analysis work. The code's development began in early 1992 as a means to answer questions and perform sensitivity studies in support of safety analyses. An impact level 2ESQ Users' Guide (WHC-SD-GN-SWD-30002) and Validation and Verification Document (WHC-SD-GN-SWD-30003) were prepared, internally reviewed by WHC, and cleared for public release in October 1993. Westinghouse Hanford Company states that the code's verification and validation was done in accordance with the WHC Quality Assurance Manual (WHC-CM-4-2) and configuration control is maintained per WHC Standard Engineering Practices (WHC-CM-6-1) and WHC SAR Work Procedures (WHC-CM-6-32). A specific evaluation of the codes verification and validation effort was not performed; however, confirmatory calculations using accepted codes were performed to validate specific results obtained using the GXQ code for PFP (see SER Section 9.0.4.2).

Various structural, soil structure, and response spectra codes were used by WHC to develop the design basis earthquake scenario. All of the PFP seismic supporting documents referenced in the FSAR have been examined, and a list of all of the computer codes used in the seismic analyses has been compiled. After discussions with WHC software custodians and URS/Blume (WHC's subcontractor for much of the seismic analysis), it was determined that all but one of the software codes had been formally verified and validated. The following is a list of software codes used in the seismic analysis section, along with the developer of each code.

<u>Computer Code</u>	<u>Developer</u>	<u>Verification Document</u>
PCSMSPC 0.0	URS/Blume	URS Ltr (See References)
ADINA 84 N14	URS/Blume	URS Ltr
AUTOPIPE 4.20/4.15	WHC	WHC-SD-GN-CSWS-30003
ANSYS PC Linear 4.3	WHC	WHC-SD-GN-CSWD-301, Rev. 0
COMPU-TEC FRAME3D	WHC	Verification Status Unknown
FLUSH	WHC	WHC-SD-GN-CS-CSWD-005
FLUSH 2.21	URS/Blume	URS Ltr
SAP IV	URS/Blume	URS Ltr
SAP 100	URS/Blume	URS Ltr
PCFLSPEC	URS/Blume	URS Ltr
ANSYS 4.4	WHC	EDT-109206

The FATHOMS®, IRRAS, GXQ codes, and the various seismic codes used by WHC in performing the safety analyses have either been through a formal verification and validation process or have been used sufficiently in the industry to assume their validity. However, COMPU-TEC FRAME3D has never been formally verified by WHC. The code was used for one scoping calculation and was not used for performing authorization basis analyses; therefore, its validation status is determined to have no effect on the safety analysis. The DOE review concludes that the computer codes used by WHC to perform safety analysis calculations have appropriate verification and validation documents associated with them.

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10.0 CONDUCT OF OPERATIONS

Described in Chapter 10.0 of the FSAR are the PFP administrative and operating activities that ensure continued and safe plant operations. The specific elements discussed are the organization structure, preoperational testing, training program and qualifications, normal operations, emergency planning, and decontamination & decommissioning.

This chapter was written in accordance with NUREG-3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants*. The information contained in this chapter meets the requirements found in the NUREG document. However, since the FSAR was released DOE has issued several orders relating to the conduct of operations; specifically, DOE Orders 5480.19, *Conduct of Operations Requirements for DOE Facilities*, and 5480.23, *Nuclear Safety Analysis Reports*. This chapter of the FSAR was not written to and does not meet the requirements in these orders relating to conduct of operations.

WHC has issued and is operating by WHC-IP-821-PUO, *Plutonium Operation Administration Manual*, which was developed to fully comply with DOE Order 5480.19. The procedures and programs developed by WHC to define the conduct of operations are adequate for the limited mission of the facility. However, this FSAR chapter should be updated to include the contents and format of the new administration manual as part of the FSAR's first annual update.

Since there were no DOE orders specific to conduct of operations before the FSAR was released, the acceptance criteria for the individual sections of this chapter are based on the new DOE Order 5480.19 and applicable portions of 5480.5, *Safety of Nuclear Facilities*. Guidelines for operator training are established the new DOE Order 5480.20, *Personnel Selection, Qualification, Training, and Staffing Requirements at DOE Reactor and Non-Reactor Nuclear Facilities*, and chapters V and XIV of DOE 5480.19. Acceptance criteria for emergency planning were based on DOE Order 5500.3, *Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for Department of Energy Operations*.

10.1 ORGANIZATIONAL STRUCTURE

Discussion. This section addresses the organization's administrative structure. It specifically defines the line of responsibility for operating the PFP facility to ensure that plant operators, engineers, managers, and smaller organizations within the facility understand their commitments. Personnel qualifications and responsibilities are also defined for managers to the technical support staff. The functions, responsibilities, and authorities of WHC management are defined in WHC-CM-1, *Management Policies*, and WHC-CM-1-2, *Organization Charts and Charters*.

Evaluation Findings. This section was reviewed based on criteria contained in DOE Order 5480.19 and paragraph 10 of DOE Order 5480.5. The scope of this review included investigations of the management organization programs and procedures used to define PFP operations. Although the present information in this section is dated (organizational structures are dated September 1990), it was determined that the information in the FSAR complies with the intent of the DOE orders and is acceptable. WHC is currently operating with WHC-IP-821-PUO, *Plutonium Operation Administration Manual*, which complies with DOE Order 5480.19. The applicable contents of this manual should be included in the FSAR as part of the first annual update.

10.2 PREOPERATIONAL TESTING AND OPERATION

Discussion. At PFP, preoperational testing is the initial phase of the overall Operational Readiness System. The Operational Readiness System includes all the steps taken to test the systems which have already operated and those operations which represent a new technology not previously applied in plant. The extent of testing and/or review is determined by the type of operation, applied technology, potential risk associated with the operation, the degree and complexity of required controls, extent of plant modifications, and the length of time since the operation was last active.

At PFP, the Operational Readiness System consists of four phases:

- (1) Acceptance testing which demonstrates that fabrication, assembly, installation, and construction requirements have been met as defined in the procurement specifications. Acceptance testing procedures are prepared and/or performed by the vendor/supplier to assure the facility custodian that all procurement requirements have been met.
- (2) Preoperational testing is performed on the equipment and the process line in preparation for operational testing. The preoperational testing procedures are prepared to assure that the system operates in the context of the actual process under real or simulated conditions.
- (3) Operational testing is prepared and conducted by facility operators against the final in-service process/ facility configuration. These tests are also known as the cold run.
- (4) The Operational Readiness Review (ORR) is the final testing phase which verifies that all elements (plant, procedures, personnel records, etc.) are in conformance with DOE requirements.

WHC-CM-1, *Management Policies*, defines the overall policy, requirements, and responsibilities of WHC in the implementation of DOE orders regarding operational readiness. WHC-CM-1-3, *Management Requirements and Procedures*, delineates the purpose, scope, definitions, responsibilities, and procedures associated with the ORR. The ORR program relies on acceptance, preoperational, and operational testing as a source of information for

determining operational readiness. The testing activities are controlled by WHC-CM-6-1, EP-4.2, *Testing Practices*.

Evaluation Findings. The testing programs (acceptance, preoperational, and operational) established by WHC which are included as part of the ORR have been reviewed. The scope of the review has included investigations of the testing programs and procedures used to define and implement these tests. The WHC acceptance, preoperational, and operational testing programs are adequate to exercise the PFP systems for determining operational readiness. The ORR will ensure that all system are operationally ready for restart. Based on the information in the FSAR and the associated WHC documents, the preoperational testing and operations leading to the ORR are acceptable.

10.3 TRAINING PROGRAMS

Discussion. The training programs for PFP personnel are designed to provide the knowledge required to operate the plant in a safe manner. The individual training programs for Plutonium Operations, PFP Engineering, and PFP Maintenance organizations are discussed. The responsibilities of the managers and training personnel are also defined. The training program for the nuclear supervisors and managers at PFP is detailed in WHC-CM-5-8, *Plutonium Finishing Plant Administration*, and WHC-WD-56110-004, *200 Area Analytical Laboratory training Program for Technologist, Managers, and Scientists*.

The training program for the Nuclear Operators at PFP is detailed in WHC-5-8, WHC-WD-56110-004, and the current Westinghouse Hanford Atomic Metalworkers Trades Council Agreement. In order for plant operators to be formally certified, they must complete a self-study package aided by subject matter experts (SME), obtain on the job training under the guidance of an OJT instructor, be evaluated and recommended by a supervisor/manager, pass a comprehensive written examination, and undergo an independent evaluation.

The PFP process engineering training program is designed to be in compliance with the employee training requirements specified in WHC policy manuals WHC-CM-8-6, *Site Support*, and WHC-CM-4-29, *Nuclear Criticality Safety*. Operational Support Services, a WHC support organization, provides the implementation requirements for the maintenance training program in WHC-CM-8-7, Section 603, *Maintenance Training Program*.

This section also addresses the frequency of recertification, other training requirements, and plant policy in regards to retraining. Also, training records are used to document the status of completed training actions. These records are used to flag when requalification training is due. Official training records for all categories of training are maintained and distributed by the cognizant organization.

Evaluation Findings. This section was reviewed based on criteria established in DOE 5480.20, and chapters V and XIV of DOE 5480.19. It has been determined that WHC follows the requirements and guidelines for operator training established in DOE 5480.20,

Personnel Selection, Qualification, Training, and Staffing Requirements at DOE Reactor and Non-Reactor Nuclear Facilities, and Chapters V and XIV of DOE 5480.19, *Conduct of Operations Requirements for DOE Facilities*. Self assessment by management, reviews of incident reports, NRC safety bulletins and notices, and DOE safety notices are placed in the required reading program and incorporated into written operating procedures. In preparing to restart the facility, WHC has emphasized training of appropriate operator responses to conditions outside of the normal operating regime as described in emergency operating procedures. (A review of operating and emergency operating procedures has determined that the procedures are adequate for obtaining the desired results.) Since the WHC training program meets the requirements and guidelines established by the DOE and incorporates a self assessment program to learn from past events, the training programs are found to be acceptable.

10.4 NORMAL OPERATIONS

Discussion. It is the policy of WHC that safe operation and maintenance of the PFP processes is maintained by strict procedure compliance and facility configuration control. Engineering configuration management ensures that current controlled operating documents are available at all work locations, affected personnel are made aware of controlled document changes, and obsolete, void, or outdated controlled documents are removed from the work locations. Administrative procedures are contained in WHC-CM-5-8, *Plutonium Finishing Plant Administration*. Engineering procedures are contained in WHC-CM-5-5, *Operations--General Administration*, and WHC-CM-6-1, *Standard Engineering Practices*. These manuals encompass all engineering activities such as plant modification, temporary modifications, design verification, etc. WHC-CM-7-5, *Environmental Compliance*, provides the standards, procedures, and recordkeeping requirements to support environmental monitoring and hazardous material activities. Individual plant procedures implement the requirements of this manual.

Operating procedures are provided for existing processes and are developed for new operations as they are required. The preparation, review, and approval of these procedures is detailed in WHC-CM-8-2 and WHC-CM-5-6. These documents also control temporary or one-time procedures which are issued as work procedures and approved work packages. Maintenance of the plant is performed according to written maintenance procedures and calibration procedures. These procedures are contained in WHC-CM-8-2, *200 Area Support Services*. Corrective maintenance is performed to preplanned and approved work packages.

Control of the procedures is maintained by document WHC-CM-5-8, Section 2.5, *Operating Procedure Review*, and procedure change authorizations. It establishes a documentation system necessary to ensure that personnel are made aware of new/revised procedures, procedure change authorizations, or other operating documents. Procedure change authorization is used to ensure that PFP personnel are aware of any changes affecting the PFP facilities or equipment before work is initiated.

The WHC records management program has been established to comply with the contractual requirements of the DOE. The program outlines the requirements of the maintenance and disposal of record and nonrecord material and is implemented by WHC-CM-5-8. This procedure defines the responsibilities and procedures for records management, assignment of retention periods, transfer of records, disposition of records, and approval and disposition of the records inventory and disposition schedule and is implemented in WHC-CM-1-3.

Occurrence reporting is in accordance with DOE Order 5000.3A, *Occurrence Reporting and Processing of Operations Information*, and is implemented in WHC-CM-1-3. Environmental reporting system requirements are derived directly from federal and state regulations. The system is implemented in WHC-CM-7-5, *Environmental Compliance*. Additional requirements for recordkeeping associated with hazardous materials is found in WHC-CM-5-16, *Hazardous Waste Management*. These two manuals contain the regulatory justification for the records produced and maintained.

Evaluation Findings. This section was reviewed based on criteria contained in DOE Order 5480.19 and paragraph 8 of DOE Order 5480.5. Also, occurrence reporting was reviewed in accordance with DOE 5000.3A, *Occurrence Reporting and Processing of Operations Information*. The scope of this review included investigations of the management organizational programs and procedures used to define and control PFP operations. The methodology used to control and modify facility procedures was also reviewed. Although the present information in this section does not meet the format requirements of DOE Order 5480.19, it was determined that the information in the FSAR and the associated WHC documents complies with the intent of the DOE orders and is, therefore, acceptable. WHC has developed WHC-IP-821-PUO, *Plutonium Operation Administration Manual*, to comply with DOE Order 5480.19. The information contained in the FSAR should be updated to include applicable material contained in this manual as part of the first FSAR annual update.

10.5 EMERGENCY PLANNING

Discussion. Westinghouse Hanford Company operates the Hanford site, including PFP, for DOE. The WHC-CM-4-1, *Westinghouse Hanford Company Emergency Plan*, was prepared in accordance with the 5500 series of DOE orders, and specifies the Hanford site emergency plan. The plan establishes the management organization and plans for managing emergencies and establishes the criteria and requirements for area emergency control and building emergency plans and procedures. The plan applies to all WHC operations, vendors, visitors, and non-WHC tenants in WHC controlled facilities. It establishes a WHC Emergency Response Organization (ERO) to manage and respond to credible emergency conditions; to request emergency assistance; and to inform WHC management, RL, and other site tenants of such events. The plan has been coordinated with RL, other RL contractors, and the state and local authorities.

The PFP emergency plan, WHC-IP-0263-234-5Z, *Building Emergency Plan for Plutonium Finishing Plant*, provides specific employee, management, and building organization

information and procedures to be used in PFP's readiness to respond to emergencies. The plan also addresses the required training to enhance and demonstrate the effectiveness of building, area, and company level emergency management and response capabilities.

Evaluation Findings. Acceptance criteria for emergency planning was based on the 5500 series of DOE orders, specifically DOE Order 5500.3, *Reactor and Nonreactor Nuclear Facility Emergency Planning, Preparedness, and Response Program for Department of Energy Operations*. The scope of this review included investigations of the PFP emergency plan and all supporting documents associated with the Hanford site emergency plan. The PFP emergency plan as presented in the FSAR does not comply with DOE Order 5500.3. However, since the issuance of the FSAR, the PFP has revised their emergency plan. This new plan, implemented by WHC-IP-0263-PFP, *Building Emergency Plan for Plutonium Finishing Plant Complex*, in conjunction with WHC site emergency plan, RL emergency plan and procedure manuals meet the intent of the requirements for facility/building emergency plans. All Washington State requirements for hazardous waste contingency plans are also met by these plans. Since the required information to demonstrate compliance with DOE orders has been provided, the new PFP emergency plan is found to be acceptable. However, WHC should include the new emergency plan details in the PFP FSAR as part of the first update. They should also continue to develop the implementing procedures to come into full compliance with the DOE 5500 series of orders.

10.6 DECONTAMINATION AND DECOMMISSIONING

Discussion. DOE Order 6430.1A, *General Design Criteria*, mandates that facilities where radioactive materials are used be designed to simplify facility decommissioning and disposal or reuse. The approach taken at PFP consists of five phases and includes the following; characterization of the facility through chemical, physical, and radiological examination, completion of the environmental review process, defined by *National Environmental Policy Act of 1969*, RCRA, and CERCLA, engineering planning, decommissioning operations, and post decommissioning.

Evaluation Findings. WHC has demonstrated compliance with the intent of DOE Order 6430.1A and therefore the D&D program as presented in the FSAR is acceptable.

11.0 OPERATIONAL SAFETY REQUIREMENTS

As a result of the accident analysis discussed in Chapter 9 of the PFP FSAR, WHC identified OSRs in WHC-SD-CP-OSR-010, *Plutonium Finishing Plant Operational Safety Requirements*. This document is evaluated in this Chapter of the SER, since it replaced Chapter 11 of the FSAR. The DOE approved the PFP OSRs in a memorandum dated May 20, 1994 (Knuth 1994b). The approval letter was based on reviews of the pending PFP FSAR, for which this SER is being written, and three supporting safety assessment documents (WHC 1994a, WHC 1994b, and WHC 1994c). The approval was also based on satisfactory resolution by WHC of previous DOE comments on early drafts of the OSR (Ford 1993). The approved OSR does not permit operation of the product concentrator, filtrate evaporator, or Glovebox MT-5 hotplates. This OSR SER has the same limitations on operability of the product concentrator, filtrate evaporator, and Glovebox MT-5 hotplate as in Knuth (1994b), namely, that they remain shutdown. Major upgrades to the OSRs from those submitted in 1991 were as follows:

- Addition of a control on plutonium inventory
- Addition of generic LCOs and surveillance requirements (SRs)
- Addition of modes to define specific conditions of the facility for LCO applicability
- Deletion of LCOs on operations no longer part of mission
- Deletion of extraneous "recovery" and "audit point" requirements in the LCOs and LCSs
- Deletion of "actions, recovery, audit point, and bases" in the Administrative Controls
- Use of 3-column format for actions and use of logical connectors
- Addition of a *Use and Application* section for acknowledging accepted practices for frequency, completion time, and notes
- Use of more precise action statements
- Removal of allowance for flammable gas bottles from LCO to Administrative Control
- Disallowance of acetylene in PFP
- A new Administrative Control on waste drum inventory
- LCOs and Administrative Controls on fissile material inventory
- Refocus of the LCOs to only those needed from the updated accident analysis and associated safety assessment documents (1991 to 1994)
- Deletion of safety limits
- Addition of Design Features (Appendix B to the OSRs)
- Change of format, content to agree with DOE Order 5480.22 and DP TSR Document of Examples (Knuth 1994a)
- Use of all capitalized defined terms throughout the OSR and especially in action statements.

Evaluation Criteria. The OSRs were evaluated by DOE using evaluation criteria developed by WHC and from DOE Orders and Standards. The evaluation criteria documents cover a substantial time range. The initially submitted OSRs (WHC 1991) were based on early DOE Orders, which only had general and comparatively nonspecific guidance. As the analysis progressed through 1994, other more current Orders and guides became available and were used. Westinghouse Hanford Company agreed to upgrade their OSRs to come into substantial compliance with the new evaluation criteria even though they only had committed to compliance with the earlier DOE Orders. The following documents were used as the basis for evaluation criteria:

- WHC threshold criteria (Chapter 9, Section 9.4 of the PFP FSAR)
- Appendix C of the OSRs, Safety Envelope Features
- DOE Order 5480.5
- DOE Order 5480.ZZ
- DOE Order 5480.22
- Draft DOE-STD-3005-93
- Draft DOE-STD-3009-YR (Gears 1994).

The OSR safety envelope is based on the WHC threshold criteria from Section 9.4 of the FSAR which establishes on-site and off-site radiological dose limits to which accident doses are compared. Such comparisons result in identification of accident preventive and mitigative equipment and non equipment controls needed to be in place in order for the accident analysis assumptions to remain valid. These preventive and mitigative equipment and non equipment controls define the safety envelope for development of OSRs. Westinghouse Hanford Company presented their methodology for determination of the safety envelope in Appendix C to the OSRs. This methodology is very similar to the guidance in draft DOE-STD-3009-YR, *Evaluation Guidelines for Accident Analysis and Safety Structures, Systems, and Components*, and is based on:

- Protection of the public
- Protection of the on-site workers
- Protection of the facility workers.

The specific radiological dose limits are presented in the FSAR, Section 9.4, *Risk Guidelines and Evaluation*. No specific radiological dose guidelines are defined for the facility worker.

Toxicological risk acceptance criteria are specified as ERPG-1 to ERPG-2 for the off-site public and ERPG-2 to ERPG-3 for the on-site worker for nominal range of frequency per year 1 to E-06. The toxicological criteria are not used for OSR determination. This is consistent with DOE Order 5480.5 and draft DOE Order 5480.ZZ. Although not consistent with DOE Order 5480.22, not using toxicological criteria for OSR development is accepted because the earlier Orders were in effect at the time of original issuance of the OSRs and because the new rule making on TSRs is expected to have the requirement for chemical hazards deleted.

The guidance in DOE Order 5480.22, draft DOE-STD-3005-93, draft DOE-STD-3009-YR, and the guidance in the TSR Document of Examples was done on a negotiated basis with WHC as these documents and guides were developed after the original OSR submittal. The major criteria from these guides have been captured in the major upgrades enumerated in the Background Section above. Other major guidance either already in place at PFP or adopted from the draft DOE-STD-3009-YR was the off-site radiological dose criteria for defining thresholds for OSRs and recognition of developing LCOs from worker safety requirements and from consideration of defense-in-depth. However, because WHC began the OSR effort before this documents were available and because of the draft status of the documents, there is not strict compliance with them.

However, WHC has made major upgrades to the OSRs to be substantially in compliance with these later Orders and guides. The areas of non compliance with regard to treatment of chemicals in the OSR, and small differences in off-site radiological dose criteria are not significant. The establishment of on-site worker radiological dose limits is a site specific commitment and is not mandatory in the Order or other DOE guidance documents.

11.1 DEFINITIONS, USE AND APPLICATION, GENERIC LCOs

Discussion. These sections of the OSR are substantially similar to those of the Defense Programs TSR Document of Examples. As Defense Programs has already approved the TSR Document of Examples and distributed it for use by Defense Programs facilities, this forms the basis for the acceptability of the subject sections. Further, even though a basis for acceptability exists, Defense Programs reviewed these sections extensively and had them changed as necessary to reflect PFP specific requirements. These sections have general information on the following:

- generic LCOs and surveillance requirements (SRs)
- modes to define specific conditions of the facility for LCO applicability
- use of 3-column format for actions and use of logical connectors
- acknowledging accepted practices for frequency, completion time, and notes
- change of format and content to agree with DOE Order 5480.22 and DP TSR Document of Examples
- use of all capitalized defined terms throughout the OSR and especially in action statements.

Most of this general information is not facility specific and is based on many years of experience. The information in the TSR Document of Examples in these areas was made applicable to DOE nonreactor facilities but was paraphrased from the guidance in the Nuclear Regulatory Commission's (NRC) recent Technical Specification Improvement Program (TSIP) from which it developed the improved Standard Technical Specifications, which for one reactor type is in NUREG 1431. Many maintenance and operations (M&O) contractors, including WHC, have developed their own versions of this general information based on their parent company's participation in the NRC's program and their involvement in the Energy

Facility Contractors Group (EFCOG).

Evaluation Findings. The three areas which involve PFP specific requirements are the definitions, modes, and generic LCOs, especially, the default action LCO 3.0.3. The three modes of operation, limited handling, and standby, and the six mode locations of main facility, plutonium reclamation facility, remote mechanical C line, plutonium processing support laboratories, analytical laboratory, and ZB complex are acceptable. They offer a compromise of keeping modes to a minimum but still retaining adequate operational flexibility and reflect independent facility process/locations. Only the definitions of "dispersible" and "recovery plan" are peculiar to PFP and are appropriately used. The default action LCO 3.0.3 requires change to a mode outside the applicability. However, all LCOs for PFP have applicability to all modes. So, there is in fact, no fall back to a default action. Westinghouse Hanford Company recognizes this and accepts that each LCO adequately addresses contingencies.

11.2 SAFETY LIMITS

Discussion. DOE-STD-3009-YR states that few, if any, Category 2 facilities are expected to have safety limits. Although DOE Order 5480.22 identifies a potential requirement for safety limits, the interpretation of this requirement in DOE-STD-3009-YR is that safety limits should be reserved only for situations which can result in acute catastrophic doses to the public, i.e. comparable to doses from reactor severe accidents, possibly ten to a hundred thousand rem. This perspective is evidenced in DOE Order 5480.22 by the requirement of full facility shutdown to the safest mode if a safety limit is violated. Plutonium Finishing Plant has a maximum off-site dose of 2.07 rem due to the seismic event (see Table 1 of SER Chapter 1). This maximum off-site dose is below the risk acceptance guideline of 25 rem (Table 1 of SER Chapter 1). Thus PFP does not have potential off-site doses in the range that would reach the threshold for establishing safety limits.

Evaluation Findings. Westinghouse Hanford Company has proposed no safety limits. This conclusion is acceptable. This acceptability is based on current DOE consideration of the need for safety limits. The WHC OSRs started with several safety limits in 1991 to reflect their interpretation of the then existing standard Draft 5480.ZZ. These safety limits were on criticality prevention, solution temperature in the laboratory concentrator, and prevention of explosions in Glovebox MT-5. DOE 5480.5 was the Order in effect at that time and was not very specific as to the requirement for safety limits. DOE challenged the basis for these 1991 safety limits and as a result, WHC developed a position based on draft versions of DOE-STD-3005-YR and DOE-STD-3009-YR. These DOE draft standards reflect interpretations based on DOE Order 5480.22. Thus WHC's safety limit position evolved to one closer to that of 5480.22 in 1994.

11.3 LIMITING CONDITIONS FOR OPERATION

As a result of the accident analysis discussed in SAR Chapter 9, WHC identified Limiting

Conditions for Operation (LCO). These Limiting Conditions for Operation are judged acceptable. The DOE reviewed and commented on several previous versions of these OSRs. Through this iterative review process covering one and half years, WHC has improved the PFP OSRs in responding to the DOE comments. The basis for the acceptability of each LCO is discussed below.

11.3.1 LCO 3.1.1 Criticality Prevention System

Discussion. This criticality prevention system LCO defines preventive equipment for inadvertent criticality and is based on the accident analysis in FSAR Section 9.2.3, *Criticality*. The following Criticality Safety Evaluation Reports (CSERs) provide specific analyses for this criticality prevention LCO: 1) CSER 78-013, addressed the 26-Inch Process Vacuum Liquid Detection Interlock System and 2) CSER 80-029, addressed glovebox Criticality Drains.

The following CSERs addressed glovebox criticality safety features required within the PFP: 1) CSER 79-018, 2) CSER 79-019, 3) CSER 79-022, 4) CSER 79-024, 5) CSER 87-005, and CSER 88-001.

The 26-Inch Process Vacuum Liquid Detection Interlock System prevents accumulation of solution in the 26-Inch Process Vacuum HEPA filters by isolating the HEPA filters and shutting down the vacuum pumps upon detection of solution upstream of the filters. Glovebox Criticality Drains prevent the accumulation of solutions above safe heights. The criticality safety features in gloveboxes and the Room 166 sump limit the solution volume in gloveboxes and sumps to analyzed values by reducing the space available to solutions within the glovebox or sump. The boron content of the Raschig Rings in the Room 166 sump also provides for absorption of neutrons to decrease reactivity.

Evaluation Findings. This LCO is judged acceptable on the basis that it adequately defines the most significant criticality prevention equipment for the protection of workers and it contributes to defense-in-depth for the protection of the off-site public. This LCO is not necessary to meet WHC Risk Acceptance Guidelines (RAGs) as a hypothetical bounding criticality of E+19 fissions results in acceptable doses on-site (0.7 rem) and off-site (0.053 rem). Fatalities are possible for facility workers from an inadvertent criticality due to the hands-on nature of certain operations. Specific facility worker doses were not calculated in the FSAR.

11.3.2 LCO 3.1.2 Criticality Detectors and Alarms

Discussion. This criticality detector and alarm system LCO defines mitigative equipment for personnel response to inadvertent criticality and is based on the accident analysis in FSAR Section 9.2.3, *Criticality*. The Criticality Detectors and Alarm system provides for mitigation by providing prompt detection of a nuclear criticality and by providing a warning to personnel for immediate evacuation. Rapid evacuation of facility personnel minimizes radiation dose. Failure of the system will not increase the probability of a criticality occurring, however, it

may result in unnecessary exposure to facility personnel located near a criticality should it occur.

Evaluation Findings. This LCO is judged acceptable on the basis that it adequately defines the most significant criticality mitigation equipment for the protection of workers. This LCO is not necessary to meet WHC evaluation guidelines as a hypothetical bounding criticality of E+19 fissions results in acceptable doses on-site (0.7 rem) and off-site (0.053 rem).

11.3.3 LCO 3.2.1 Filtered Exhaust from zones 3 and 4

Discussion. The uncontrolled spread of airborne contamination outside of the facility is prevented by a combination of physical barriers and the building ventilation systems. The probability of contamination release to the environment is minimized by maintaining progressively decreasing pressures from non-contaminated zones to the highly contaminated gloveboxes and process cells. This pressure differential results in the exhaust air from these areas flowing through High Efficiency Particulate Air (HEPA) filters before release to the environment. Similarly, air from rooms which contain gloveboxes is drawn through HEPA filters before release to the environment. Integrity of the HEPA filters is ensured by the specified periodic surveillances.

Evaluation Findings. This LCO is judged acceptable on the basis that it adequately defines the most significant mitigative equipment for the protection of on-site workers and protection of the off-site public. The accident analyses provided in chapter 9 of the FSAR assume that each stage of HEPA filtration required by this LCO will have a Decontamination Factor (DF) of 2000 (equivalent to a particulate removal efficiency of 99.95%). This LCO provides mitigation for all the accidents in Chapter 9 except the seismic event, waste drum fire, criticality, nitric acid release, and the carbon tetrachloride release, because either no credit is taken for the filters, or they occur outside the building or their releases are not affected by the filters (fission gases, nitric acid, and carbon tetrachloride releases). The FSAR did not determine whether this level of filtration may not be needed to ensure that off-site and on-site exposures following internal accidents or upsets in PFP buildings remain below the Risk Acceptance Guidelines in FSAR Section 9.4. However, until safety analyses using less restrictive particulate removal efficiencies have been completed, system particulate removal efficiencies of 99.95% are required for each PFP HEPA filtration stage covered by this LCO.

Even if this LCO on filtered exhaust were determined to be not needed based on meeting the on-site or off-site dose criteria of FSAR Section 9.4, this LCO should be retained because it provides for the most significant, active defense-in-depth mitigative feature at PFP.

11.3.4 LCO 3.2.2 Effluent Stack Air Monitors and Alarms

Discussion. The effluent stack monitors and alarms provide for defense-in-depth as they may indicate a failure of the HEPA filters or ventilation system. A variety of events, such as liquid and powder spills, pressurized containers and glovebox breaches, ventilation upsets, and

several explosion related events are considered in FSAR Sections 9.1, *Abnormal Operations*, and 9.2, *Accidents*. Each of these scenarios involves a short term condition which could challenge the PFP Ventilation Confinement barrier and result in a plutonium release to the environment through a stack. The requirement for effluent stack air monitors and alarms helps to monitor the operability of the ventilation system HEPA filters and to detect and identify the source path of a potential radioactive release. The monitoring equipment also helps to ensure that the plant operates within regulatory requirements for releases to the public.

The surveillance SR 3.2.2.2 verifies the alarm setpoint is less than 135 cpm. The alarm setpoint is based on conservative implementation of DOE Order 5484.1 for occurrences requiring investigation and WHC-CM-7-5, *Environmental Compliance Manual*. The basis for the alarm setpoint of using the requirements from WHC-CM-7-5 provides considerable assurance that no release above the Risk Acceptance Guidelines would go undetected.

The Effluent Stack Air Monitors and Alarms ensure prompt detection of elevated plutonium concentrations in effluent air. The system consists of one or more alpha continuous air monitors (CAMs) at each stack, a vacuum source for each CAM, and an alarm network to annunciate equipment failures and plutonium levels above the setpoint.

Evaluation Findings. This LCO is accepted because it establishes a significant defense-in-depth feature of the FSAR for the protection of on-site workers and protection of the off-site public. This LCO is not necessary to meet WHC evaluation guidelines as it does not directly prevent or mitigate either on-site or off-site doses from the analyzed accidents. Failure of the Effluent Stack Air Monitors and Alarms system may cause a delay in alerting facility management to a plutonium release to the environment. However, an abnormal condition or accident coupled with a HEPA filter failure would have to occur before any potential for exceeding WHC's risk acceptance guidelines could exist.

11.3.5 LCO 3.2.3 Supply Ventilation Seismic Shutdown

Discussion. The dose consequences to the public and on-site workers is based on the design basis accident described in Section 9.2.4 of the FSAR, which is a 0.20 g earthquake with a mean annual occurrence of 10^{-4} .

Plutonium is released within and outside the 234-5Z Building as a result of the seismic event. Significant internal building structural and component damage may occur.

Primary physical barriers challenged are:

- Building structure (walls, roof, ceilings, and floors)
- Gloveboxes
- Ventilation system.

FSAR Section 9.2.4 presents the accepted risk envelope for the seismic event. This scenario is based on the seismic shutdown system functioning.

The consequence of the supply ventilation system seismic shutdown system not performing as designed is that supply fans may continue running after a seismic event. If, in addition, confinement barriers are breached and exhaust fans fail, the 234-5Z Building and 236-Z Building could be pressurized causing a significant on-site release of respirable plutonium. For the FSAR, the assumption is made that the supply fans will be shutdown by the seismic switch.

The supply ventilation system seismic shutdown requirement is to shut down the 234-5Z supply fans in the event of a greater than 0.07 g earthquake.

The supply ventilation system seismic shutdown is designed to shut down the 234-5Z Building supply and dry air process fans (supply fans) upon occurrence of a seismic event (earthquake). Two accelerometers are installed that sense the occurrence of a seismic event. The accelerometers are calibrated to respond to an earthquake equal to or greater than 0.07 g and send a signal to the system shutdown circuitry, which in turn, shuts down the 234-5Z supply fans. Only one accelerometer and its associated shutdown channel is required for system operability.

Evaluation Findings. This LCO is accepted because it mitigates the consequences to the on-site worker and the public during a seismic event. This LCO is considered necessary to meet WHC evaluation guidelines for the on-site worker and the public since it is assumed to shutdown the supply ventilation system. No dose has been calculated with the supply ventilation system running during a seismic event.

11.3.6 LCO 3.3.1 Control of Dispersible Plutonium

Discussion. The accident releases are developed from detailed glovebox to room release scenarios based on individual glovebox inventories of dispersible plutonium. Throughout Sections 9.1 and 9.2 of the FSAR there are discussions of abnormal events and accidents which result in potential plutonium releases from the PFP. The most significant releases occur as a result of an earthquake.

Safety analyses contained in WHC-SD-CP-SAR-021 consider numerous parameters in the development of plutonium releases from PFP buildings during accidents. Parameters include location within the building, structural OSR Design Features, seismic resistance of gloveboxes and filters, chemical and physical form of plutonium, and the mass of plutonium present. The mass of plutonium potentially present is based on existing residual quantities, on masses allowed by Criticality Prevention Specifications, and additional restrictions of Table 5.22.2 in AC 5.22. Inventories of dispersible plutonium outside of gloveboxes are included in these totals. The amounts of plutonium designated and used in the analyses represent an upper bound on the potential consequence of accidents and the resulting risk. To provide an upper

bound of the risk, the calculations used plutonium powders to model plutonium dispersion in all areas where plutonium solids are present to maximize the potential dispersion. Using this approach in the analysis negates any need for controls on physical form of the plutonium inventory. The use of plutonium powders in the scenarios and conservative release fractions assures that an upper bound of the risk is identified and that the control on mass alone is sufficient to stay within the calculated risk envelope. In order to maintain the facility within the accepted risk envelope, control of plutonium mass in PFP is required.

Evaluation Findings. This LCO is accepted because it establishes inventory control of fissile material and, therefore, is the basis for all radiological accident source terms (except for the waste drum fire which is covered by AC 5.24). It is the basis for meeting radiological acceptance criteria for both the on-site worker and the off-site public.

The Buildings 234-5Z, 236-Z, 2736-ZB repackaging glovebox, and analytical laboratory plutonium inventory specifications of this LCO are considered adequate to provide a top level control. The total plutonium inventory allowed by this LCO is consistent with that for the seismic analysis used in the FSAR.

Although it is recognized that individual glovebox inventory control is required to maintain accident analysis off-site dose consequences, it has been judged that level of detail is more appropriate for the Administrative Controls, and can be found in AC 5.22. That level of detail presented in AC 5.22, of glovebox by glovebox inventory specification, is not appropriate for an LCO as it would defeat the original intent of LCOs, which is to focus the operator's attention on a manageable number of the most significant safety requirements.

11.3.7 LCO 3.4.1 Product Concentrator Temperature Control

Evaluation Findings. This LCO is accepted because it establishes that the product concentrator be maintained shutdown. This is appropriate since the FSAR does not authorize operation of the product concentrator. However, this LCO may be deleted in the first annual update of the FSAR, since this accident scenario has been omitted from the FSAR.

11.3.8 LCO 3.4.2 Filtrate Evaporator Temperature Control

Evaluation Findings. This LCO is accepted because it establishes that the filtrate evaporator be maintained shutdown. This is appropriate since the FSAR does not authorize operation of the filtrate evaporator. However, this LCO may be deleted in the first annual update of the FSAR, since this accident scenario has been omitted from the FSAR.

11.3.9 LCO 3.4.3 Miscellaneous Treatment Glovebox 5 Temperature Control

Evaluation Findings. This LCO is unnecessary and may be deleted in the first annual update of the FSAR. This is appropriate since a determination was made that an explosion in MT-5 was incredible and that the seismically-induced deflagration resulted in very low on-site and

off-site dose consequences.

11.4 ADMINISTRATIVE CONTROLS

Discussion. This section has listed top level requirements summarized from facility broad based, extensive programs and requirements. These programs and requirements are in the following areas:

- Contractor Responsibility
- Compliance
- Operational Safety Requirement Violations
- Occurrence Reporting
- Revisions to the Operational Safety Requirements
- Minimum Operational Shift Complement
- Facility Change Control
- Unreviewed Safety Questions
- Personnel Qualification and Training
- Facility Reviews and Audits
- Audit Records Requirements
- Nuclear Criticality Safety
- Radiation Protection
- Effluent Monitoring and Sampling
- MODE Changes
- OSR Interfaces with other Facilities
- Fire Protection
- Flammable Gas Inventory
- Plutonium Inventory
- OSR Compliance Program
- Management of Transuranic Material Outside Facility Buildings.

Most of this information is not facility specific and is based on many years of experience. The information in the TSR Document of Examples in these areas was made specifically applicable to DOE nonreactor facilities but was founded from the guidance in the Nuclear Regulatory Commission's (NRC) recent Technical Specification Improvement Program (TSIP) from which it developed the improved Standard Technical Specifications, which for one reactor type is in NUREG 1431. Many M&O contractors, including WHC, have developed their own versions of this general information based on their parent company's participation in the NRC's program and their involvement in the Energy Facility Contractors Group (EFCOG).

The flammable gas inventory, plutonium inventory, and management of transuranic material Administrative Controls are unique to PFP. In each of these areas, WHC evaluated the hazards and wrote a corresponding safety assessment document (SAD). As a result of the evaluations in the SADs, the limits in the Administrative Controls were developed. The limit

of no more six one pound propane bottles in AC 5.21 is from the flammable gas SAD, WHC-SD-CP-SAD-003, *Safety Assessment for Flammable Gas Explosion at PFP*. The glovebox by glovebox plutonium inventories in AC 5.22 are from the inventory sensitivity SAD, WHC-SD-CP-SAD-002, *Safety Assessment for PFP Inventory Sensitivity of the Seismic Event*. The 15 gram transuranic limit in a single 55 gallon waste drum in AC 5.24 is from the waste drum fire SAD, WHC-SD-CP-SAD-007, *Safety Assessment for Fire in TRU Waste Stored Outside of PFP*. In each case, these are more restrictive limits than have previously been permitted. The accident analyses in the SADs have been incorporated into Chapter 9 of the FSAR.

Evaluation Findings. This Section of the OSR is substantially similar to those of the Defense Programs TSR Document of Examples. As Defense Programs has already approved the TSR Document of Examples and distributed it for use by Defense Programs facilities, this forms the basis for the acceptability of the subject section. Further, even though a basis for acceptability exists, Defense Programs reviewed this section extensively and had changes made as necessary to reflect PFP specific requirements.

11.5 BASES (Appendix A of the OSRs)

Evaluation Findings. The Bases are not part of the OSR enforceable requirements and as such, are acceptable. Again, the basis information, as presented in format and required subject content, follows the guidance in the TSR Document of Examples. And, therefore, for these aspects, has its acceptability basis from the Defense Programs endorsement of the TSR Document of Examples. And again, the guidance in the TSR Document of Examples is patterned from the NRC program and carries with it experience lessons from many years of commercial reactor operations. The specific technical content in the bases, aside from format and subject content, is specific to PFP and has been judged adequate. Although not part of the enforceable OSR requirements, the OSR Bases are subject to change control per AC 5.7, *Operational Safety Requirements Basis control*.

11.6 DESIGN FEATURES (Appendices B and C of the OSRs)

Discussion. The PFP SSCs that are important to safety, collectively referred to as the Safety Envelope (SE) Features, are discussed implicitly in the various Chapters of the PFP Final Safety Analysis Report (FSAR), particularly, in the "Accident Analysis" Section of Chapter 9 and explicitly in Operational Safety Requirements (OSRs). Other documents that address the SE Features include Criticality Safety Evaluation Reports (CSERs).

Design Features are that subset of the Safety Envelope discussed in Appendix C which are passive features and if altered or modified, could adversely affect facility safety. These Design Features shall maintain their function, such as confinement, compartmentation, structural integrity, and function consistent with the concept described in accident analyses to prevent and mitigate, confine, or stabilize as appropriate.

Three of the eight preventive or mitigative systems in Appendix C are considered to provide the functions necessary to minimize the risks associated with events identified in the FSAR and be classified as Design Features:

Feature - Structural confinement boundaries necessary for prevention or mitigation of releases of radioactive materials outside the facility. These features prevent or mitigate the effects of accidents such as those caused by earthquakes, strong wind, missiles of external origin (e.g., wind driven), explosions and fire. Portions of PFP building structures provide this Design Feature. Supporting Document WHC-SD-CP-SDD-007, *Definition and Means of Maintaining the Structural Confinement Portion of the PFP Safety Envelope*, provides detailed definition of this portion of the PFP Safety Envelope. This document identifies specific features such as walls, columns and steel structural members which provide this Design Feature.

Feature - Ventilation confinement boundaries necessary for prevention or mitigation of releases of radioactive materials outside the facility. These features prevent or mitigate releases of plutonium and other radioactive particulate contamination in the building ventilation exhaust stream above safe levels. Supporting Document WHC-SD-CP-SDD-005, *Definition and Means of Maintaining the Ventilation System Confinement Portion of the PFP Safety Envelope*, provides detailed definition of this portion of the PFP Safety Envelope. This document identifies specific filters and related hardware which provide this Design Feature.

Feature - Equipment installed to provide an essential barrier utilized in the double contingency principle for prevention of nuclear criticality. The double contingency principle (reference AC 5.15) requires that operations be controlled such that at least two unlikely independent, and concurrent changes must occur before the margin of sub-criticality required by the WHC Nuclear Criticality Safety Program (WHC 1988) can be lost. Where practicable, one of these changes involves equipment design. Specific facility features which provide one of the essential barriers applied in the double contingency principle for criticality prevention are identified in WHC-SD-CP-SDD-008, *Definition and Means of Maintaining the Criticality Prevention Design Features Portion of the PFP Safety Envelope*.

Evaluation Findings. The Design Features of Appendices B and C describe in detail those features not covered elsewhere in the Operational Safety Requirements (OSRs) that, if altered or modified, could have a significant effect on safety. These Design Features of the OSR are acceptable. The acceptance criteria is that these Design Features adequately meet the intent of DOE Order 5480.22 for Design Features.

11.7 REFERENCES

U.S. Department of Energy (DOE). 1994. "Evaluation Guidelines for Accident Analysis and Safety Structures, Systems, and Components," DOE-STD-3005-YR, February 25, 1994.

U.S. Department of Energy (DOE). 1986. "Safety of Nuclear Facilities." DOE Order 5480.5.

U.S. Department of Energy (DOE). 1992a. "Technical Safety Requirements." Draft DOE Order 5480.ZZ.

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Gears, G. 1994. Memorandum to Distribution dated June 30, 1994, "Informational Transmittal of DOE-STD-3009, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports."

Knuth, D.F. 1994a. Memorandum to Distribution dated June 23, 1994, "Document of Example Technical Safety Requirements."

Knuth, D.F. 1994b. Memorandum to DOE-RL dated May 20, 1994, "Approval of Plutonium Finishing Plant Operational Safety Requirements."

Mecca, J.E. 1994. DOE-RL letter to J.A. Ford dated April 29, 1994, "Plutonium Finishing Plant Operational Safety Requirements, WHC-SD-CP-OSR-010."

NUREG-1431. *Standard Technical Specifications, Westinghouse Plants*. Issued by Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, September 1992.

WHC 1988. *Nuclear Criticality Safety Manual*. WHC-CM-4-29, Rev. 0. Westinghouse Hanford Company, Richland, Washington.

WHC 1991. *Plutonium Finishing Plant Final Safety Analysis Report*. WHC-SD-CP-SAR-021. Westinghouse Hanford Company, Richland, Washington.

WHC 1994a. *Safety Assessment for PFP Inventory Sensitivity of the Seismic Event*. WHC-SD-CP-SAD-002. Westinghouse Hanford Company, Richland, Washington.

WHC 1994b. *Safety Assessment for Flammable Gas Explosion at PFP*. WHC-SD-CP-SAD-003. Westinghouse Hanford Company, Richland, Washington.

WHC 1994c. *Safety Assessment for Fire in TRU Waste Stored Outside of PFP*. WHC-SD-CP-SAD-007. Westinghouse Hanford Company, Richland, Washington.

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12.0 QUALITY ASSURANCE

Described in Chapter 12.0 of the FSAR is the facility's Quality Assurance (QA) program. Department of Energy site contractors are required to develop QA programs in accordance with DOE Orders. Over the last several years, contractors for RL have developed QA Programs to meet RL Order 5700.1A, which was RL's interpretation of DOE-HQ Order 5700.6A. In September 1986, DOE Order 5700.6B was issued that superseded 5700.6A—no RL order was issued for this DOE-HQ order. The common trend of these DOE orders is to apply the American Society of Mechanical Engineers (ASME) standard NQA-1 *Quality Assurance Program for Nuclear Facilities*; some exceptions were made for previously DOE approved QA Programs, and some additional requirements and considerations were included in the regional order.

On August 21, 1991, DOE-HQ issued DOE Order 5700.6C. This order not only supersedes DOE Order 5700.6B, but also requires implementing activities not covered by either NQA-1 or specified by the previous DOE orders (e.g., Quality Improvement). It is important to note that the current interpretation of this order is for the regional DOE offices to issue 5700.6C directly to the site contractors.

The review of this chapter focused specifically on requirements in DOE Order 5700.6B which was in effect at the time the FSAR was issued. A cursory review was made of United States Nuclear Regulatory Commission (USNRC) Regulatory Guide 3.39, *Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants* (which is referenced in section 12 of the subject FSAR), primarily the QA section Chapter 12. The review did not include the appendices, non-mandatory guidance to NQA-1, or any DOE orders concerning FSARs (e.g., DOE Order 5480.23, *Nuclear Safety Analysis Reports*).

12.1 DISCUSSION

Specifically addressed in Chapter 12.0 of the FSAR are the following topics:

- Organization - Describes the Quality Assurance (QA) organization within WHC, and establishes the responsibility and authority for various groups in the organization. Discusses location and freedom of the WHC QA organization to assess the systematic implementation of requirements by all WHC organizations.
- Quality Assurance Program - Discusses the various organizations involved in implementing the QA program and the training necessary to indoctrinate personnel into the program. Discusses the graded application of NQA-1 requirements and provides a reference for structures, systems, and components covered by the WHC QA process and establishes the requirement for periodic management review.

- Design Control - Establishes operational and maintenance controls required to maintain the plant within the design configuration specified by the FSAR. Also, this section establishes the controls necessary to verify design input and changes.
- Procurement Document Control - Cites the WHC's QA manual for control of documents related to procurement of services and materials.
- Instructions, Procedures, and Drawings - Describes how instructions, procedures, and drawings are prepared, issued, controlled, and maintained.
- Document Control - Discusses documents that are to be controlled, and other WHC publications that provide guidance in regards to document control systems.
- Control of Purchased Items and Services - Establishes guidelines to ensure that purchases conform to procurement document requirements.
- Identification and Control of Items - Describes a system to identify and control materials, parts, and components that will prevent the use of incorrect or defective items.
- Control of Processes - Defines responsibilities for developing procedures and performing special processes, including welding and brazing. Requires qualification of personnel and adherence to codes and other standards.
- Inspection - Discusses an inspection program required to identify characteristics, methods, and acceptance criteria and provide for recording objective evidence of inspection results.
- Test Control - Describes the measures for ensuring the satisfactory performance of equipment or system components in service.
- Control of Measuring and Test Equipment - Describes the measures taken to ensure that measuring and test equipment used in PFP activities are properly identified, controlled, adjusted, and calibrated at specified intervals to maintain accuracy within prescribed limits.
- Handling, Storage, and Shipping - Describes the measures necessary to establish control over the handling, shipping, cleaning, and preservation of materials, equipment, and end products to prevent damaged or deterioration.
- Inspection, Test, and Operating Status - Describes measures taken to identify the status of inspection and test activities, either on items or in documents traceable to items, to ensure that only items which are installed, operated, used, or processed have passed the required inspections, tests, and laboratory analyses that confirm acceptability.

- Control of Nonconforming Items - Describes the measures taken to identify items that do not conform to specified requirements, so as to prevent their inadvertent installation or use.
- Corrective Action - Discusses policy in regards to corrective action and states that conditions adverse to quality will be identified promptly and corrected as soon as practical.
- Quality Assurance Records - Requires the specification, preparation, and maintenance of records which furnish documentary evidence.
- Audits - Requires that planned and scheduled audits be performed to verify compliance with all aspects of the QA program and to determine its effectiveness.

12.2 EVALUATION FINDINGS

Although the FSAR was developed prior to issuance of DOE Order 5700.6C, *Quality Assurance*, it now falls under its scope. While NQA-1 addresses many of the requirements of 5700.6C, some requirements contained in 5700.6C are outside those contained in NQA-1 (e.g., Continuous Improvement) to which Chapter 12.0 of the FSAR was written. In general, Chapter 12.0 meets the requirements of NQA-1 and is found to be acceptable with the following exceptions:

- Supplement 8S-1, paragraphs 3.2 *Limited Life Items*, and 3.3 *Maintaining Identification of Stored Items*, of NQA-1 are not discussed in Section 12.8.
- Supplement 10S-1, paragraph 5.2, *Combined Inspection and Monitoring*, of NQA-1 is not clearly addressed within Section 12.10.
- Supplement 11S-2, for computer program testing is not covered in Section 12.11.
- The requirements of Supplement 12S-1, paragraphs 3.2, *Control*, 4, *Handling and Storage*, and 5, *Records*, are not addressed by Section 12.12.
- Basic Requirement 14 of NQA-1, the requirement for "The authority for application and removal of tags, markings, labels, and stamps shall be specified" is not clearly addressed within the FSAR.
- The responsibility and authority requirement of NQA-1 Supplement 15S-1, paragraph 4.2 is not addressed by Section 12.15 of the FSAR.
- Within section 12.17 of the FSAR, it appears that the requirement for retrievability contained in NQA-1 Basic Requirement 17 is not addressed. Also, the requirements for single, alternate, and dual storage facilities and the definitions for "Lifetime" and

"Nonpermanent Records" and retrieval and disposition of records contained in Supplement 17S-1 of NQA-1 are not adequately addressed.

These inadequacies are not considered to be significant safety issues, since credit is not taken for their existence in the development of accident scenarios and abnormal operations as presented in Chapter 9.0, *Accident Safety Analysis*. Furthermore, a plan to implement the new requirements in, and fully comply with, DOE Order 5700.6C was prepared by WHC and submitted to DOE in May, 1992. Westinghouse Hanford Company should make revisions to Chapter 12.0, *Quality Assurance*, during the PFP FSAR's next annual update to fully comply with the new requirements established in DOE Order 5700.6C.

APPENDIX A

MASTER LIST OF RESPONSES TO TECHNICAL QUESTIONS REGARDING THE PLUTONIUM FINISHING PLANT'S FINAL SAFETY ANALYSIS REPORT

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APPENDIX A

MASTER LIST OF RESPONSES TO TECHNICAL QUESTIONS REGARDING THE PLUTONIUM FINISHING PLANT'S FINAL SAFETY ANALYSIS REPORT (DRAFT 1991)

A.1 GENERAL SAFETY REVIEW

1. What efforts are made to learn from past events to preclude the reoccurrence of operator actions which lead to abnormal or accident situations? Are these efforts a part of the mandatory operator training program?

Response. This is a global issue central to the conduct of operations at DOE facilities but not specific to the PFP FSAR. At PFP there are several mechanisms to manage corrective actions relative to control of operator actions. Self assessment by supervision, reviews of incident reports, NRC safety bulletins, DOE Safety Notices, and other similar advisories channel pertinent information into the required reading program and written operating procedures which must be adhered to by operating and support personnel. In preparation of restart of the PFP operator training includes emphasis on appropriate operator responses to abnormal conditions outside the normal conditions described in written operating procedures. Subject Matter Experts are utilized in this training. In a meeting with PNL the definition of Subject Matter Expert was discussed. While there is no documented definition, the choice of a Subject Matter Expert is made by management based on experience on a case by case basis. When the training personnel feel that they need help explaining some phase of work they call the manager who is responsible for the subject area and ask that a knowledgeable person be sent to the training area to help. We depend upon the judgement of our management team to pick the proper person.

2. An operational safety requirement (Section 11.2.2) is established for the solution temperature in the laboratory waste concentrator that has a safety limit requirement of 133°C and a limiting control setting of 125°C. Neither of these temperature limits is discussed in the accident scenario for the laboratory waste concentrator (Section 9.2.10). Please provide a discussion of the temperature controls that monitor both the safety limit and the limiting control setting temperatures. Have these temperature monitors been evaluated for common failure models and, if so, do any common mode failures exist? How fast of a response time is needed to prevent the solution temperature to go from 125 to

133°C if the limiting control setting temperature monitor does not shut down the heating system during the most limiting abnormal event? Please discuss these temperature limits and controls and their impact on the probabilities and consequences of the most limiting accidents for the laboratory waste concentrator.

Response. This system currently does not have temperature monitoring or controls. In the past temperatures have been controlled by steam pressures alone. The lack of direct monitoring of the parameter of interest and control based on that information has been recognized by Westinghouse and system modifications are currently underway. The steam valves to the concentrator are locked out of service until modifications to the system are completed. This concentrator will not be used in the upcoming resumption of PRF operations.

The accident analysis for the concentrator will be revised and the FSAR updated prior to concentrator start up.

- 3. What modifications to the facility that were credited in the safety analysis report will not be implemented? Please discuss why these modifications are not necessary.**

Response. As part of the PFP Restart Program currently being implemented by Westinghouse the safety authorization basis contained in the FSAR is being specifically addressed and validated as appropriately implemented through physical facility inspections and procedural control reviews. Should any modifications assumed in the FSAR not be completed prior to restart they will be evaluated relative to the safety basis of the FSAR and justification documented as not necessary. Modifications which are necessary to remain within the safety envelope but not implemented would constitute an Unreviewed Safety Question and would be managed and disposition with the DOE. Such conditions might result in the need to revise the authorization basis contained in the FSAR.

- 4. A number of reference sections were referred to by other sections in the document. However, these sections were not found. For example:**

- Section 9.2.4.6.3.2B was referenced by Section 9.2.1C.2, but it is absent.
- Section 9.2.4.10 was referenced by Section 9.2.4.1, but it is absent.
- Section 9.2.4.6 was reference by Section 9.2.1c.4, but it is absent.

*** **Response.** Section 9.2.4 was extensively reorganized shortly before publication. A number of references to paragraphs were missed. The designations in question should be as follows:

Section 9.2.4.6.3.2B	is	9.2.4A.6.3.2.B
Section 9.2.4.10	is	9.2.4.3.2
Section 9.2.4.6	is	9.2.4B.2.2

Westinghouse will correct this problem in the next FSAR update since the conclusions concerning risk are not affected.

5.a Has a loss of instrument air system been evaluated as an accident initiator or to assess the impact on safety systems' response to other accident initiators?

NEW Response. Although instrument air is essential for continuity of normal operations the loss of instrument air is not considered an accident initiator. Control dampers and valves are designed and installed to fail in a predetermined safe position, e.g., either open or closed, upon loss of instrument air. The failure positions chosen allow the maintenance of adequate ventilation air flows during abnormal conditions such as loss of electricity or instrument air. The consequences of ventilation and safety system component failures are analyzed in sections 9.1.7 and 9.1.8 for noxious chemicals and radioactive materials.

As part of the activities currently ongoing to assure operational readiness and implementation of the PFP FSAR, the accident scenarios and abnormal operational conditions described in Chapter 9 are being reviewed to assure that proper configuration and operational controls are in place for all systems, structures and components important to safety of the facility. The extensiveness of this review has been discussed with Pacific Northwest Laboratory personnel and a copy of the detailed criteria has been provided to them for information. These criteria specify requirements for operational and configuration controls for facility systems which are categorized as systems, structures or components important to safety. The identification process for such items includes consideration as to whether the item provides a support or actuation function that enables a system, structure or component important to safety to function successfully. Items such as power supplies, vacuum, air, cooling systems, instrumentation and controls are systematically considered. Safety system boundaries and allowable outage times for maintenance are identified by this process. This review provides the assurance that items considered at a system level in the FSAR do not have their performance impacted by support services or components.

During several discussions with PNL reviewers concerns were raised as to the classification of certain administrative actions and equipment systems relative to not being included in the Operational Safety Requirements of Chapter 11 and the Safety Classification of certain systems. Such items included ash loading for the roof, room

continuous air monitors, and seismic shutdown of ventilation exhaust fans. The process described above is a detailed component specific evaluation of the role the items play in the accident and abnormal events analyses contained in Chapter 9 of the FSAR. The evaluations which are drafted and undergoing internal review have addressed these issues as facility safety envelope items which do not have specific OSRs but have a configuration controlling OSR which deals with Design Features.

5.b Has gradual degradation of instrument air system components been considered such that component performance may not meet performance requirements on demand?

Response. The facility has several levels of redundancy and interties between the process air system and the instrument air system such that a continued supply of instrument air is readily assured. Several compressors which have operated since the plant was built were replaced in the last couple of years. Distribution piping can be easily replaced should significant degradation occur. Additionally, generic failure rate data was conservatively applied by the analysts with the purpose of accounting for a variety of ill-defined phenomena that could degrade performance.

5.c What is the basis for failure rates used for the critical performance of instrument air system components and what records support use of these failure rates?

Response. The basis for failure rate data accompanies the use of the data in the text of each accident presented. Component failure mode and rate data has not been either generated or kept at the PFP.

6. Has the potential for a loss of power to the PFP other than for the seismic event been considered, evaluated, and documented, and, if so, where?

Response. The FSAR documents function of the building ventilation system on loss of electrical power. Loss of electricity is examined as one of the failures for each accident analyzed where appropriate. Westinghouse found no energy sources that require electricity to keep them in a safe configuration. To the contrary, Westinghouse found that generally loss of electricity enhances safety by removing an energy source. For example, there is an emergency shutdown button on the RMC-line control panel and at the exit door to shut off all electricity to the line should rapid evacuation be required.

7. Section 9.0.1.2.4.4 should evaluate and substantiate that the PFP maintenance policies and practices support the modeling of equipment and failure rates used in this facility safety analysis report rather than state this is an assumption.

Response. The sentence in question conveys the message that poor maintenance or different equipment or facility configurations would negate the validity of the analyses.

This type of situation is implicit in any analysis and was stated for clarification of the perspective from which the analyst proceeded in the analysis. There are no data upon which to compare modeling and policies/practices, even if it could be made in a quantitative manner. Maintenance of a reasonably good conduct of operations is an underlying assumption in every safety analysis even those containing quantified data. Westinghouse believes that the use of human factors failure rate data is sufficiently conservative to allow for degradation in conduct of operations. Current emphasis by WHC on conduct of operations and maintenance including requirements of DOE Order 4330.4A makes such degradation very unlikely. In the context of this comment no amount of detailed equipment evaluations to support the failure rates used in the FSAR would change the truth of the statement which may be restated in general terms, "If conditions are different from that assumed in the analysis, the analysis is not valid."

8. **Section 9.0.2.4 credits PFP staff with taking protective actions under flood or heavy rainfall conditions. Are these actions proceduralized? If not, how than they be credited? Also, the evaluation takes credit for apparent drainage routes for a potential 11-inch rain fall within 24 hours. Please demonstrate that drainage and operator actions that are credited are based on design and established procedures.**

Response. The 11 inch rainfall discussed is an incredible (10^{-6} per year) event. Planning for such events is not proceduralized. That is one of the reasons why a shift surveillance manager is always on duty. The manager can direct emergency responses and call in additional help where and when needed. Emergencies of much less importance require a call to the emergency response center on 811 and managers are well acquainted with these general requirements. It is noteworthy that the facility was in existence and survived the maximum rainfall recorded in the last 80 years without damage.

9. **The PFP facility safety analysis report should include one section for transportation accidents within the site and one section for concerns about tornados, although the probability of these events occurring are very low they need to be addressed in more detail.**

Response. Transportation accidents are outside the scope of a facility FSAR. To and from the Hanford Site, transportation of radioactive and hazardous materials are regulated by commercial transportation requirements. In the case of onsite transportation of radioactive materials separate Safety Analysis Reports for Packaging are prepared to demonstrate comparable compliance with DOT requirements. Such controls provide acceptable risk from transportation accidents. Within the PFP fenced yard vehicles move much more slowly than permissible highway speeds and traffic is sparse at best. Transportation accidents within the yard area were dismissed as inconsequential by the analysts and not documented along with the thousands of other such evaluations made during the preparation of the FSAR.

As stated on page 9-30 of the FSAR, the Hanford Site Standard Design Criteria have been revised to exclude tornados as a site consideration and consequently they are not addressed.

- 10.a The table on Page 9-404, Section 9.2.4a.2.11.E, does not appear to be consistent and is based on a single reference. The table indicates that Glove Box HC-4 has an area of 25,300 cm² and a critical depth of 10 cm. Therefore, this should result in a critical volume of 253 L, not 256 L, as listed in the table. Why is there a discrepancy in the volumes?**

Response. The table on 9-404, Section 9.2.4a.2.11.E, does indeed have some errors based upon interpretation of the data from two sources and conversion from inches to centimeters. The effect does not alter the conclusions of the section, and should be addressed in the next FSAR revision. Glovebox HC-4 was defined in the reference letter as being critical at 4 inches depth. When this depth was converted to centimeters, it was rounded off to 10 vs. 10.16 cm.

The calculated volume, using 10.16 vs 10.0 cm gives the higher volume, stated in the table. The letter erred, however, in interpreting the maximum subcritical depth derived in the CSAR addendum. The reactivity at 4.0 inches was calculated, but it was above the normal 0.95 $k_{\text{effective}}$ value, so interpolation was used to find a value of 3.81 inches.

- 10.b How is it possible that Glove Box HC-227s, with a lower surface area, has a shorter critical height than Glove Box HC-4?**

Response. The CSAR addendum for HC-227S calculated reactivity for 3.03 and 3.53 inch depths and stated the maximum subcritical was higher than 3.5 inches. If extrapolation were performed, a value closer to the 3.8 to 4.0 would have been correct, thus both gloveboxes should have a common depth in the table (3.8 inches/9.65 centimeters).

Since the conclusions are unchanged and there is no impact on safety, Westinghouse Hanford Company will defer revision of this information until the next general update of the FSAR.

- 11. For Section 9.2.1A, 1B, and 1D, describe a) the specific safety systems for the process, and b) laboratory evaporators/concentrators. Are there safety systems beyond limiting the solution temperature? Are there safety systems for spills, fire, and explosions? Are there redundant safety systems? If so, please list. List procedural safety systems separately from hardware systems.**

Response. Please refer to the response to Question 2 above which deals with this same subject. The laboratory concentrator will not be used in the upcoming PRF operations. Chapter 9 of the FSAR describes in detail the accident scenarios involving the process

concentrators including the systems and components relied upon for safety. The prevention of explosions in these concentrators is prevented by redundant pressure and temperature controls as well as operating procedures which implement the safety requirements for these systems documented in an Operating Specification Document. A copy of the applicable Operating Specification Document has been provided to Pacific Northwest Laboratory personnel.

12. **Have all possible mechanisms for age-related degradation of process and safety equipment been analyzed, e.g., stress-related corrosion and fatigue? Have any nondestructive tests been performed to evaluate age-related degradation in the permeation reduction factor, RMC, or storage facility? The age of the process and facility varies between 40 (1949) and 25 (1964) years, excluding recent replacements and refurbishments. Have the empirical failure rate data and tests been compared to those assumed in the SAR and has this been documented?**

Response. No, all possible mechanisms for age related degradation have not been evaluated. Limited nondestructive tests have been performed on components where degradation would have potential for significant concerns such as the stainless steel lining of the Plutonium Reclamation Facility process cell floor. The process for evaluating Unreviewed Safety Questions (USQ) at PFP would identify significant findings of failure data as an USQ if data invalidate assumptions used in the FSAR. As discussed in the response to Question 5 a review of systems, structures and components important to safety in being conducted to assure proper implementation of the FSAR prior to restart of the Plutonium Reclamation Facility. The review process criteria provides for the establishment of a surveillance program to assure functional testing and calibration of safety equipment and retention of historical records of these activities. These records will enable assessments of the effects of aging to be made in the future. The preventive maintenance, functional tests and calibrations are included on the schedule and plans for restart of the facility. These tasks are being upgraded to comply with the requirements of DOE Order 4330.4A.

13. **What specific hardware safety systems have been installed to mitigate the accident scenarios having unacceptable dose consequences (Paragraph 2.5.1 on Page 2-22), specifically:**

- 2.4.1.1, Explosion in MT-5
- 2.4.1.4, Explosion at HC-13MD?

The SAR only mentions systems which are being developed. Are there redundant hardware systems? Please discuss the procedural systems as a separate item.

Response. The temperature control equipment necessary to mitigate the MT-5 explosion as described in Chapter 9 has been installed subsequent to preparation of the FSAR. With this new equipment and procedures the accident probability was analyzed

in Chapter 9 and determined to be incredible. The HC-13MD glovebox operation is associated with plutonium metal production will no longer be done at PFP. This is a moot question.

14. **Is the dissolver system in Room 170 included as a part of the SAR? Is it is similar to MT-5?**

Response. The ash dissolution process in glovebox HC-46 located in Room 170 is specifically excluded as stated in the Introduction to the FSAR on page 1-2.

A.2 CRITICALITY

- 1.a **Is tank TK-42 a geometrically favorable tank for criticality control?**

Response. Vessels Tk-42 and Tk-43 are associated with the $\text{Pu}(\text{NO}_3)_4$ Concentration System in PRF and are both geometrically favorable for criticality safety.

Vessel Tk-42 is a 4-barrel stainless steel bank tank with each barrel 5 inches in diameter.

Vessel Tk-43 is the "falling-film evaporator" which is a titanium pipe (4 inches in diameter) with an external steam jacket (6 inches in diameter).

- 1.b **Are the glass Raschig rings in TK-42 used for criticality control?**

Response. TK-42 is the CCP feed tank, it doesn't contain any Raschig rings. TK-43, the CCP stripper evaporator described above, is packed with titanium Raschig rings in the 4 inch pipe for heat and mass transfer (steam stripping of entrained/dissolved organic).

- 1.c **If they are, how is the poison content of the rings and the volume of the rings verified?**

Response. The Raschig rings are used for process reasons and not for criticality safety, therefore, no verification is required.

- 2.a **The criticality accident and radioactive release is based on the Nuclear Regulatory Commission's Regulatory Guide 3.35. The Regulatory Guide makes specific assumptions about both the number of fissions and the aerosol formed during the criticality accident. Will the wide variety of plutonium isotopics at PFP impact these assumptions? Specifically, address the dynamics of the fission bursts in plutonium that has much lower or higher plutonium-240 isotopics.**

Response. The Pu isotopic makeup at PFP will have minimal effect upon the criticality dynamics, the shutdown mechanisms postulated, and the aerosol releases from a criticality. Most scenarios for a criticality for PRF Restart are for moderated systems in which ^{239}Pu is the predominant Pu isotope involved. Very high ^{240}Pu material would have ^{241}Pu also present, but the fission dynamics and shutdown mechanisms would not be significantly different for ^{241}Pu than ^{239}Pu . In fact, even a fast system would behave similarly whether it was initiated by only ^{239}Pu or all the isotopes. The aerosol from the criticalities would have Pu isotopes present in a ratio similar to the originating distribution rather than only ^{239}Pu , but the overall effect and all fission products ratios would remain the same. This is of minor significance for the PRF FSAR since no criticality has offsite effects.

The dominant shutdown mechanisms for solution criticality accidents are microbubble formation, density changes due to heating, and in some cases the dispersal of fissile material in the immediate vicinity of the accident (splashing). Secondary effects can include the accumulation of fission products that act as neutron absorbers, and gradual changes in fissile concentration through heating. All of these effects are essentially independent of the isotopic composition of the fissioning material. By contrast, the estimate of the number of fissions that could be expected in a criticality accident is based on a conservative evaluation of the historic accidents. This approximation is an order-of-magnitude estimate, and although accepted by subject matter experts as a conservative value for most systems, the uncertainties associated with the estimate are large compared to the minute effects of fissile composition, distribution, or isotopics.

2.b If so, will molten plutonium be present during currently planned operations and how does this impact releases if exposed to air or water?

Response. The RMC Line (Pu Metal Production) will be operated in the future to produce only PuO_2 , and during PRF Restart Operation, no Pu-metal buttons will be reprocessed through PRF associated gloveboxes, so molten Pu reactions with water and/or air should not be a concern.

3.a What are the qualifications of the Criticality Safety Representative?

Response. The PFP Criticality Safety Representative meets the qualifications which are defined in the Westinghouse Hanford Company Nuclear Criticality Safety Manual, WHC-CM-4-29, Nuclear Criticality Safety Manual, Section 3.0, Training, which is excerpted below:

The Criticality Safety Representative must successfully complete the managers and engineers criticality safety training course. For fissionable material facilities, the process engineers phase II certification program should be completed, if available. The expertise, topics and depth of required knowledge are appropriate to the

complexity of the criticality safety program in the facility, and are subject to review by the independent safety review organization. Criticality Safety Representative qualification is completed by examination by a qualification board. Board member(s) will represent the independent safety review organization, Criticality Engineering Analysis, process engineering, the operating organization, and may include others as agreed by the board. The 2 year requalification process may be satisfied by one of the following:

1. A formal CSR retraining class.
2. An oral requalification examination by the qualification board.
3. A requalification evaluation based on an appraisal of the CSRs work conducted by the qualification board.
4. A waiver of the requalification process by the qualification board (on a case-by-case basis).

The CSR is required to read and demonstrate knowledge of the content, as part of the FSO requalification, the portions of the facility SAR relevant to criticality safety, WHC-CM-4-29, and the appropriate CSERs.

3.b Do the Criticality Safety Representatives report directly to the plant manager or what is the line management between the Criticality Safety Representatives and the plant manager?

Response. At the PFP the CSR reports to the Manager, PFP Facility and Operations Assurance and Training, who in turn, directly reports to the Manager, PFP. Currently there are two designated backup Criticality Safety Representatives. One reports directly to the manager, PFP Engineering who reports to the plant manager and the other reports directly to the plant manager.

4.a Criticality analysis requires two unlikely events to occur before criticality is possible. An unlikely event is defined as an event that is not expected to occur in the lifetime of the facility (reference Section 6.1.3.1.1). Has the facility ever suffered a violation of a criticality imposed administrative spacing limit or a spill of powder? If the facility has had a spacing violation or powder spill, how can spacing violations or powder spills be considered an unlikely event?

Response. The unlikely event is a set of limit nonconformance(s) and extraordinary situations, both allowed and not allowed, so that the loss of control over the combination of such conditions is defined as an event which meets the test of being unlikely. "Contingency", "unlikely event" and "worst foreseeable event" are interchangeable terms used to define the collection of conditions. Often two or more criticality safety limits must be established to block the occurrence of one "unlikely event". A nonconformance with one limit such as not conforming to spacing limits or spilling powder which has occurred would rarely result in the loss of a contingency,

both because the magnitude of the nonconformance was less extreme than assumed, and because one or more of the additional conditions associated with the contingency case was lacking. Even nonconformance with two limits simultaneously could be insufficient to result in the loss of the first contingency.

4.b If one spacing or spill has occurred, how can multiple violations be considered unlikely?

Response. The multiple nonconformances must have no identifiable common mode or common cause. An overbatch and a loss of spacing control are examples of two events that are normally independent. Multiple independent nonconformances with the same limit are far less likely than a single nonconformance, except when caused by a single error. For example, if one container could be placed in the center or on top of a cluster of properly positioned containers, that event would be a single error for purposes of the analysis. The likelihood of two independent events occurring at the same time and in the same place is obtained by multiplying the probability of the two events individually. Thus a combination of events could meet the definition of unlikely, even if some constituent events may occur occasionally.

4.c Also, if the facility has had spacing violation powder spills, how will this impact the existing criticality safety evaluations?

Response. Scenarios are developed relative to their likelihood of happening, type of barriers, and the degree of abnormality in the scenario (a 1 gram overbatch vs double-batch, or one container too close/touching another vs all containers clustered in a glovebox), and take into consideration engineered and administrative restraints. These scenarios are discussed and evaluated by engineering, operations and safety personnel. At a minimum, the criticality safety analyst, the criticality analyst-reviewer, the manager of Criticality Engineering Analysis, the PFP Criticality Safety Representative and the independent safety review organization must agree that the contingencies meet the definition of unlikely.

The development of the set of conditions considered unlikely under the definition is guided by a description of the process, the environment of the process, a list of standard error conditions that should be considered, the experience of the analyst and reviewer, and often the experience of the process engineer, the criticality safety representative, independent safety reviewers, and management. The process may be iterative.

4.d If the answers to the above are affirmative, perhaps the criticality safety requirements need to be changed. Please discuss further.

Response. Although the answers are negative, an additional discussion is offered. The scenarios postulated as "the worst foreseeable event" must result in a calculated

system reactivity that is acceptably low. That is, some additional error, accident, or equipment malfunction must occur before a criticality accident is even possible. These events must be unlikely, independent and concurrent with the first set of problems.

- 4.e **In the RMC line, significant amounts of powder exist. If powder spills have occurred in the past, how can powder spills and flooding be considered two unlikely events?**

Response. Whether a scenario is assumed to occur once in 5 years, once in 20 years or once in 100 years does not automatically exclude it from being considered as an accident or within the "worst foreseeable" cluster of events. Scenarios which have excessive calculated reactivity for the postulated "worst foreseeable condition" must have additional barriers in place to prevent the scenarios from occurring, so that collectively the loss of control on all the barriers is unlikely under the definition.

- 5.a **Are all waste streams assayed before shipment to the 241-Z tanks?**

Response. Yes. All waste streams are sampled and analyzed prior to discharge to the 241-Z tanks.

- 5.b **How are the total contents of the 241-Z tanks monitored to limit the total loading to 400 g of plutonium?**

Response. The contents of the 241-Z tanks are controlled primarily by administrative procedures. Transfers are a batch process and are restricted to 100 grams Pu or less per batch, as determined by sample analyses. Occasionally concentrations must be inferred from vessel specific gravity measurements and flowsheet knowledge. Although many solution vessels have had 450 gram per liter limits imposed on them, the vessel itself is probably safe for significantly more than this and it may relate to either shipping container restrictions or just a convenient maximum operating value. Actual flowsheet concentrations are usually sufficiently below the stated limit value that a major process upset, with other indicators, must occur prior to reaching the limit value.

- 5.c **How does the monitoring system determine the residual plutonium content of the tanks considering that the buildup may preferentially occur in the bottom corner of the tanks due to the action of the agitation device?**

Response. After neutralization, there is no good way to sample/analyze the Pu-bearing slurry, so it is kept in suspension and transferred expeditiously to minimize Pu accumulation in a 241-Z tank. As part of criticality safety program implementation, non destructive assay (NDA) of the 241-Z tank(s) and associated vault is done routinely (with multiple measurement positions within the vault), prior to reaching a 400 gram Pu input value since the previous NDA. The agitated/unagitated scenarios

discussed for the large tanks in 241-Z and in PUREX are of importance only for the multi-kilogram quantities permitted in safe acidic solutions.

- 6.a Can an acid stream be introduced into the 241-Z tanks that will cause a precipitate?**

Response. Precipitation of plutonium in solids is done intentionally in the 241-Z tanks so the accidental formation of precipitates is not of concern. Plutonium precipitates are generally caused by addition of basic solution to an acidic Pu-bearing solution, although oxalic acid, sulfuric acid, phosphoric acid would each precipitate Pu if present in high enough concentrations. There are no direct sources for addition of concentrated sulfuric, phosphoric or oxalic acid to the 241-Z tanks.

- 6.b How does the critical mass change in the 241-Z tanks if the plutonium suspension is forced to the bottom corner of the tanks by the action of the agitators instead of forming a uniform layer in the tanks?**

Response. The mass permitted in 241-Z is only 400 grams because of possible precipitation into a hemispherical or conic shape of optimum Pu concentration. This idealized shape could be critical with 800 to 1100 grams Pu. Such a shape is little affected by agitation or flatness of the tank floor. A torus or conic section formed by accumulations at the tank edges would require more plutonium for criticality than the cases assumed.

- 7. Many of the glove boxes in the PFP use a weir system to contain spills of plutonium and plutonium solutions in critically favorable geometries. During an earthquake, can the glove boxes tilt or partially collapse causing each section of the glove box to hold significantly less plutonium-bearing liquid than expected? Can this cause the lowest section to collect significantly more material than expected? Will the additional collected material cause a criticality? If the glove boxes can tilt or partially collapse, would this tilting allow fluids, suspensions, or liquids to flow from one glove box to another collecting more material than expected? During an earthquake, plutonium-bearing liquids collected on the floor of the glove boxes in the weir system will form waves and slosh. Can this cause the lowest section to collect significantly more material than expected? Will the additional collected material cause a criticality? When the liquid stacks up in a wave, can a criticality occur?**

Response. There are three types of gloveboxes to be considered in answer to these questions.

Type 1 represents the RMC metal line conveyor glovebox with the associated wet or dry gloveboxes. These gloveboxes do not have weir systems.

Type 2 glovebox is the short transfer glovebox associated with Miscellaneous Treatment processes, with 5 gloveboxes attached. Small mass limits are established for the transfer glovebox itself. The transfer glovebox does not represent a hazard because the unit mass transfers are containerized, and small total quantities are permitted. The individual attached gloveboxes have been evaluated for earthquake associated criticality scenarios, and were found to be within risk acceptance guidelines.

Type 3 is a PRF access glovebox. The access glovebox is firmly attached to the 2 feet thick, reinforced concrete canyon wall, restricting the tilt that could result even from a severe earthquake. The back wall of the glovebox is stainless steel, with welded stainless steel sleeves for piping and equipment access, anchored in the concrete wall. These sleeves are solidly attached to the glovebox for contamination control, and also serve to unite the wall and the glovebox into a single structural unit, acting as anchor-bolts for the glovebox. A longitudinal tilt down the length of the glovebox is not credible because of the structural support of the concrete canyon wall and the wall penetrations (WHC-SD-CP-SA-007, E. C. Ocoma, "PFP Safety Systems & Components Seismic Evaluation-Phase II", April 10, 1990)

If solution were present during an earthquake, there could be some sloshing. But since the glove box could not tear away from the wall and tilt more than a few inches downward from the canyon wall into the occupied space the pooled solution would not become a criticality hazard. Weirs are 2 inches deep, but have a maximum expected solution depth of 1/4 inch during operation. The weirs normally cascade to the drain section, so that solution depth in the lowest section would be limited by the 1 inch space below the criticality drain. If the glovebox tilts, the solution height in any one section still could reach an overflow depth of just over 2 inches, although at a smaller volume. This is not sufficient depth for criticality to occur. Since a longitudinal collapse is not credible, a successive cascade of solution from many sections would only occur in the direction of the criticality drain, in the normal and expected way.

While it is possible to imagine water flowing into the glovebox and collecting, this accumulation would not occur quickly. Since the earthquake would last (on the order of) seconds, and the water would accumulate for minutes (at least) before a pool was formed, the earthquake could not be a mechanism for stirring, or producing waves in the solution in excess of the 2" depth. If a strong after-shock were to occur, a wave form could be generated, but the presence of weirs, piping, equipment, and debris would dampen or attenuate the waves much as a breakwater would do. Ripples would present no hazard to criticality.

Each criticality drain would be expected to remain functional, due to the 3-inch diameter drain cross section, the screen across the glovebox width to maintain adequate solution flow rates, and the housekeeping restraints upon rags, gloves, etc. which could plug the criticality drains.

The plutonium concentration in the glovebox solution would not be expected to be as high as the near optimum reactivity range of 100 grams per liter, based upon the fact that the vessels and associated piping within the PRF canyon should have equal failure rate potentials and there is relatively little solution routinely present at high Pu concentrations.

Collectively, a solution arrangement causing a criticality in an access glovebox from an earthquake is not credible with existing controls and practices.

8. **Section 9.2.4a.2.11.E indicated that there is not sufficient solution volume available to make either Glove Box HC-4 or HC-227 critical. If an earthquake can breach a glove box window, cause a plutonium solution leak, and activate the sprinkler system, how can the conclusion of insufficient solution to achieve a criticality in either HC-4 or HC-227s be justified?**

Response. The Gloveboxes HC-4 and HC-227S, have criticality drains and fire fighting water in their respective rooms. Normally the liquid level on the glovebox floor is minimal and no leaks exist. The scenario suggested in this question would require an earthquake to occur, breaking a "ceiling" window in the glovebox but not to break any other glovebox window, port, or seam, to start a room sprinkler which coincidentally directs water at the broken window, and to rupture one or more vessels containing Pu solution, causing the solution to leak to the glovebox floor. However, even if all of these things occurred, the criticality drain could still function allowing sufficient liquid to escape, precluding a criticality accident. This type of accident was considered in sizing the criticality drains in these gloveboxes.

- 9.a **During the vault flooding, would a small drain, less than the water addition rate, cause the floating containers to collect around the drain?**

Response. No, the scenario considering vault flooding, which includes Pu containers floating to a drain appears to be precluded by design. The scuppers which provide room drainage are large enough to drain the room of fire sprinkler system discharges. Room 2 of 2736-Z is the only vault with sprinklers and small containers. The small containers are constrained by chains within the cubicles and could not float out of their positions even if completely submerged. It is possible given water of sufficient depth to float the larger drums which are in storage.

- 9.b **Would the containers collected around the drain be more than one deep and more reactive than the assumed uniform layer of cans?**

Response. No, Westinghouse has been unable to identify a mechanism which would simultaneously float containers to the drain and then stack them more than one layer deep. We cannot identify a plausible mechanism for floating containers into a whirlpool caused by an inadequate drain.

A.3 PROBABILITIES OF ACCIDENT EVENTS

1. Please evaluate the failure rates and model differences for the filtrate evaporator rather than argue that its fault tree is enveloped by the product concentrator tree evaluated in Section 9.2.1A (Section 9.2.1B.1, Page 9-225). The specific failure rates for filtrate evaporator components may be sufficiently different from those of the product concentrator, such that the minimum cut-sets for the filtrate evaporator produce a larger failure probability.

Response. The SAR stated that the probability of an explosion in the filtrate evaporator was bounded by the probability of explosion in the product concentrator because its minimum order cutset required at least two more failures than the product concentrator minimum order cutset. This argument was used because the filtrate evaporator tree was very large and difficult to quantify by hand or by IRRAS. Unless the filtrate evaporator tree is quantified, it cannot be shown that its total probability is bounded by the product concentrator tree. It is possible to show that the total probability of an explosion in the filtrate evaporator is less than 10^{-6} per year and therefore incredible. This was the original intent of the authors of this section of the SAR.

Conservative values of failure rates were assigned to the various events described in the tree. The fault tree was solved using CAFTA, a fault tree development and analysis code currently used by WHC. The failure rates used in the analysis are shown below:

<u>EVENT</u> <u>GENERIC DATABASE</u>	<u>ASSUMED PROBABILITY</u>	<u>HANFORD</u>
Process Conditions	1.0	unknown
Human Error	1E-01	1.0E-02 to 1.0E-03

<u>EVENT</u> <u>GENERIC DATABASE</u>	<u>ASSUMED PROBABILITY</u>	<u>HANFORD</u>
Transmitter	1E-02	2.6E-03/yr
Recorder	1E-01	approx. 7.0E-02/yr (SRP)
Limit Switch	1E-04	3.0E-05
Alarm	1E-05	2.3E-06
Pressure Indicator	1E-02	1.0E-02/yr
Valves	1E-03	2.0E-03 to 1.0E-05
Controller	1E-02	6.0E-03/yr
Convertor	1E-01	9.0E-03/yr
Line Plug	1E-04	4.0E-06/yr-ft
Temperature Element	0.5	0.35/yr
Relay	1E-04	7.6E-05

In calculating the failure probabilities, failure rates were taken from the Hanford Generic Database, a compilation of failure rates chosen (usually conservatively) from a variety of other well-established databases such as Savannah River, EG&G, IREP and IEEE-500. For components with an hourly failure rate, a period of one year was assumed between testing.

The fault tree was quantified with a truncation limit of $1.0\text{E-}13$. There were 37,384 resulting cutsets with a min-cut upper bound (total failure probability) of $1.96\text{E-}07$ which is defined as incredible. It is therefore clear that even with conservative numbers the scenario of an explosion in the filtrate evaporator deserves no further analysis and can be ignored with respect to consequences.

2. **The conclusion statement of Section 9.1.7A.1.2.1, Page 9-101, should describe the probabilities associated with the multiple events to substantiate that the resultant probability is low.**

Response. Specific quantification of the probabilities associated with the multiple events would be highly subjective. In order to get material out of the building from the rooms delineated one has to combine an uncontrolled pressurization with a failure of outside wall panels that were shown by tests to have a significant margin of strength. A reference (Giller 1989b), shown on page 9-106, indicates that the wall panels on 234-5Z will withstand a pressure of greater than 13.6 in. H_2O which is 150% of the force due to a 90 mi/h wind. Failure of Damper B gives the highest pressure at 1 in. H_2O . At the same time there would have to be a significant amount of respirable sized plutonium material outside of hoods, gloveboxes, ducts, or filter - boxes and the material would have to be of such a nature that it was resuspendable. Examination of the inventories shown for the effected rooms in Tables 8.2.-5 and 8.2.-6 when combined with the spill scenarios presented in Section 9.1 develop an overall aspect of low probability. The implied probabilities are based on those rule-of-thumb values for human error and either active or passive component failures generally used in the risk analysis community. The sheer number of independent, simultaneous events that would have to take place drives the overall probability to a low value even if the individual events had relatively high probabilities.

3. **The statement that glove box pressurization from gas sources is not likely should be quantified, rather than saying that it is not likely because it requires at least two failures. If any two or more potential failure sources that could pressurize glove boxes have high probability, then the resultant probability of a glove box pressurization is not low. Additionally the potential consequences of a worst case glove box pressurization should be evaluated to assess the impact of a release from these causes, even if it is a low probability.**

Response. The point being made by the author appears to have been missed by the reviewer. The "not likely" words on page 9-55 refer to "a significant release into the room" not the probability of the event. The consequences of such a breach of containment are discussed in the two preceding sections which deal with breach of glovebox containment. Including the same consequence statement in this paragraph would have made the paragraph better. Detailed discussions pertinent to this subject are also provided in Section 9.0.2.4 beginning on page 9-30. Items c.2.2.2 on page 9-47, D-19 on page 9-49 and E-8 on page 9-54 show that pressurization of the type

defined do not present a significant release due both to the nature of the material potentially available for suspension and design of the equipment.

4. **The description of ventilation system upsets evaluated in Section 9.1.7 provides a qualitative assessment of postulated sequences, determines the resulting room air pressures quantitatively, and discusses the margin of protection qualitatively. This evaluation concludes that the combined likelihood of the upset condition and the resulting low consequences of any release allow ventilation system upset conditions to be dropped from further evaluation. These scenarios should be modeled using basic event probabilities from a current source determined to be appropriate for the PFP failures as modeled, i.e., consider failure definition, success criteria, mission time, and uncertainty. This degree of detail is necessary to clearly establish the quantitative probabilities of release for comprehensive worst case ventilation upset conditions.**

Response: Please see the response to item 2 above. In addition, note that none of the other kinds of upsets besides pressurization discussed above cause plutonium to escape from primary containment.

5. **Section 9.0.1.2.4.4 should provide a description of the general approach and techniques used to determine failure rates for calculating probabilities. This description should explain failure definitions and success criteria, including mission times, and how these factors are considered in selecting failure data from the referenced sources. The techniques used to address uncertainty associated with failure rate data should also be described. Methods such as those used in NUREG-1150 should be used and discussed in the facility safety analysis report.**

NEW Response. Section 9.0.1.2.4.4 is an introductory section. It is more appropriate to discuss the requested information in the specific scenarios, as was done in Section 9.2. The general approach was to find a data base that included the item and to verify that the data base source was representative component usage to the PFP usage. The failure modes for the item that will make it typically unavailable or mispositioned are used and defined by the specific analysis being conducted.

Maintenance outages are considered as shown in the individual fault trees. For the most part there is no need to consider this factor as the equipment under consideration, such as a dissolver pot, is always shut down for maintenance. The service times indicated in the trees is so optimistic that normal amounts of maintenance time will actually reduce the stated service time and reduce the annual probability of an explosion. Element D-4 on page 9-318 gives the probability that the duct-blocking valve would be out of service when a fire started.

The concept of a success criteria and its associated failure mode is normally limited to a system. The reference source does not define system success or failure, but most

reliability data bases define the component failure mode for which they are supplying a failure rate. Finally, the mission time used to form a lambda tau calculation is based on the requirements for the system to function and the detecting of a component failure by instrumentation or test. Reasonable data bases give error bounds with the data supplied. As a rule of thumb from WASH-1400, an error factor of 3 is used for a component for which there is sufficient failure data to feel confident in the rate, 10 for questionable data, and 30 for not very well known data.

6. **The arguments presented on Pages 9-527 and 9-528 should include the source, basis, and evaluation of the probabilities to support the contention that the consequences are within acceptable limits. Similarly, the argument provided in Section 9.2.4.3.2 needs to be expanded based on quantitative numbers.**

Response: The arguments on Pages 9-527 and 9-528 are speculation. The author is suggesting that a seismically qualified controller, if installed would have reliability similar to "normal" equipment. A scan of a number of data bases shows that in general such equipment has failure rates of between 10^{-2} to 10^{-4} . Until the equipment is installed one can not cite a specific reference.

The discussion stating that the people could be evacuated the dose would not occur is incontrovertible. There was no probabilistic argument presented.

The argument presented in Section 9.2.4.3.2 is not possible to quantify or it would have been. It is a qualitative evaluation of the degree of conservatism included in the model. Additional information could have been included. For instance, the duct containing the principle amount of plutonium is a section only a few feet long located down-stream from a caustic scrubber. The nature of the duct work in the Incinerator Building, 232-Z, is presented at the bottom of Page 9-488. It is described as being schedule 10 stainless steel pipe. About half being 6 in. diameter and half 4 in. diameter. It is more resistant to crushing than typical industrial or home type ventilation duct work. Most likely material in this duct is in the form of a tightly adhering hard crust. The powder on which this was modeled was dry, ball-milled UO_2 powder. It was suspended at a height of either one or three meters in a beaker at the center of a large circular room equipped with many samplers. The beaker was quickly inverted remotely so that the powder freely fell to the floor. A realistic scenario for the pipe would probably only have it knocked down with bends and crimps as well as some limited sized ruptures with any contents falling out as pieces much above respirable size at floor level. Because we can not prove such a sequence we felt compelled to conservatively use the falling powder model.

Although we believe that agglomeration would provide a significant source term reduction it was not included. No quantitative credit can be taken for the process of agglomeration that would be taking place in the presence of a dense cloud of concrete

dust because the quantity of dust would be unknown and because the building would be open to the air and thus cloud density could not be quantified.

7. **The source of data for the parameters "C" and "N" on Page 9-563 should be indicated and the application of the equation to the PFP should be clearly explained. For example, does the value of "C" vary depending on-site proximity to air paths and does the value of "N" include the commuter air traffic to SEA-TAC airport?**

Response. The source of the data for parameter C was the NTSB and FAA data for the five year period from 1983 through 1987 (See page 9-561 of SAR). With respect to parameter N, the same page of the SAR, 9-561, indicates that there are 100 flights per year in the vicinity of the PFP area. The same page indicates that only 8% of these are actually in the area or 8 flights. The 50 used for parameter N is an extremely conservative number. Parameter C itself is a generic rate that does not include the air paths. However, calculations of probabilities for the various flight paths are given in the Table on page 9-564. The locations of the flight pathways with respect to PFP are discussed on page 9-555 and are shown on the map on page 9-556.

Subsequent discussions with the PNL reviewers involved questions regarding details of calculating probabilities of Hanford Patrol helicopter crashes. Discussion is no longer germane since the Hanford Patrol no longer has any helicopters and this scenario will be eliminated from the FSAR in future revisions.

8. **Cases where personal communication with the cognizant process engineer is cited as the basis for an input, value, or modeling decision, such as in Section 9.2.1A.1, should be backed up with an appropriate calculation or by data that is referenced.**

Response. A typical instance where data was obtained from a cognizant process engineer and used in the PFP SAR is listed on page 9-199. In this instance no other data was available for determining how long it would take to flood the CC column with organic. This is a typical example where process engineers are used as data sources. The human failure date used for this event makes it appear that a column flood is a single event That could be overlooked by a single person. A column flood is a gradually developing condition that exhibits a number of abnormal characteristics (Section 6.1.1.2.2) A flood of the C-column that would put significant quantities of organic into the product stream represents a sever upset that would have provided numerous clues to the Supervisor and Process Engineer as well as to the Operator. The operating procedure forbids tap-off of product during any stage of flood condition. There has been so little operating time in the last 2 decades that we could not find anyone who had experienced such a severe upset. The frequency estimates are based on a hypothesis extrapolated from minor upset experience, not data.

9. **The failure rates used for various components and human actions, as described on Pages 9-203 through 9-209, should consider more current sources than Dexter (1980) and NRC (1975a), and should be updated for plant specific experiences where the value is significant to the outcome particularly when Dexter (1980) and NRC (1975a) failure rates and probabilities are less conservative than current plant experience. The use of an error factor of 10 for hostile environment may be appropriate, but the reference for this practice should be given.**

Response. Site specific data is not maintained at Hanford. Therefore, failure rates based on current plant experience cannot be calculated. This forces the analyst to use generic data. The generic data used was Dexter (1980) and NRC (1975). The comment suggests using more current data. There are supposedly more current data bases. However, do they significantly differ from the data bases used? WHC maintains not. Most data bases begin with the WASH-1400 data and claim to add more recent data to it using Bayes or some other method. The basic failure rates are very close to the WASH-1400 data. The addition does not seem to do much. When using Bayes, there is the opportunity to Bayes the same basic data multiple time and bias the data. WASH-1400 was created to analyze commercial nuclear power plants. Many times in the analysis of process plants, WASH-1400 does not have data available to use on the non reactor components. Dexter has turned out to be an excellent source for such data. Dexter is based on process plants at Savannah River where the process contains radiated materials. This is not normally true of newer data bases based on the chemical process business.

10. **Please provide justification for the valving error rate used ($1.6E-4$), and also model and evaluate the potential for system upset to substantiate that there are no on-site or off-site consequences (Sections 9.0.1.2.4.2 and 9.0.1.2.4.1).**

Response. The valving error rates were developed as the result of study done over a several month period of operation. It is based on the number of reported misroutings divided by the number of valve operations made. Non-violent system upsets have no way to release significant amounts of material outside the building. Those upsets with potential for breaching the containment or confinement barriers are delineated in Section 9.2 and not in the introduction, Section 9.0.

The documents regarding this data were not published as formal WHC documents. The documentation of the basis for this data was provided by WHC to the PNL reviewers.

- 11.a **Where are the sources for probabilities stated in Section 9.2.1C.3 documented?**

Response. The reference noted as (Harker 1990) contains this information and has been provided to the PNL reviewer.

- 11.b For example, what is the source for EF1 and EF3 failure probabilities listed on Pages 9-244 and 9-245?**

Response. The failure probabilities EF1, EF2, and EF3 are all human errors of omission and are drawn from the same source as the failure probability of event EC4, which is the reference cited as (NRC 1975a), Table III, 6-1. Westinghouse will add the reference to this part of the SAR at the next FSAR update.

- 12. Section 9.2.1E.5 should include: the probability that the operator may not notice or react even if it is the first bottle, the probability of a regulator malfunction, the controls used to ensure that bottle contents are within specified limits, and a fault tree for these failure contributors.**

Response. In the future the RMC line will not be used to make plutonium metal buttons. It will generate only plutonium oxide when it is restarted due to the new mission of PFP. This eliminates the need for hydrogen fluoride gas in the operation of the process line or elsewhere in PFP. This eliminates the potential for the accident scenario described and evaluated in section 9.2.1E.

Westinghouse will remove this discussion entirely in the next FSAR update.

- 13.a The probability of acetylene use in the building should consider input from crafts and technicians in addition to the "office" engineering managers since the crafts and technicians may be more representative of actual conditions in the facility than the engineering managers. The determination of the room size probability component should consider the rooms which are potential users of acetylene. Please discuss these issues further.**

Response. The majority of craft managers interviewed about the possibility of acetylene accidents are ex-technicians and ex-crafts people themselves and are well acquainted with the use patterns in the facility.

- 13.b The probability of a tip-over due to bumping, cleaning, interference, etc., should be considered.**

Response. With regard to probabilities of tip-over and use of acetylene in specific rooms, no data is kept and the estimate given is meant to envelope such situations.

- 14. The arguments used in Sections 9.2.1G.4 and 9.2.1G.8 should provide the probabilities and their sources to support the arguments made in these two sections.**

Response. As stated in the text, it is impossible to provide a probability for an event with no identified mechanism of occurrence. As a matter of principle protection

should be provided to systems that involve hydrogen gas as a stoichiometric concentration of hydrogen and oxygen can be ignited even by the electrical effects of a nearby lightning storm. Conversations with an expert on hydrogen explosions convinced us that protection should be provided even though no conceivable mechanism was present. A detonation of hydrogen off the end of the discharge pipe in the process cell falls into the same realm of the unknown and is shown to be inconsequential. The flowsheet value for hydrogen release rates (Figure 6.1-9) is 0.33 ft³/min for 5 min (9.2.1G.7) for a total discharge of 1.7 ft³ into a cell of 43,000 ft³ (9.2.1G.8) ventilated at a rate of about 2,200 ft³/min. The discharged gas would mix and diffuse at a short distance from the point of discharge and be diluted to below a detonatable and somewhat farther out to below even an ignitable concentration.

15. **Please provide the basis for the values given for E-1 and E-2 and describe more clearly the D-7 element (Page 9-323).**

Response. The documented bases are as stated. The values of events E-1 and E-2 are engineering estimates. Event D-7 describes the probability of personnel failing to follow procedures regarding filter replacement following a filter exposure to HF gas. Engineering estimates are based on the experience and judgement of the engineering personnel who operate the plant in consultation with the author and have been concurred with by a number of reviewers.

It should be understood that even if the probabilities of D-7, E-1, E-2 and E-3 are set to 1.0 the resulting total failure probability would increase to no higher than 6.5E-08. The events in question therefore have no bearing on the integrity of the safety envelope. It is also important to note that HF will no longer be used in the PFP processes as plutonium metal will no longer be produced.

16. **Please provide a calculation or reference for the probabilities rather than only discussion (Section 9.2.2A.6).**

Response. The number of places and conditions under which "a fire" could occur approaches infinity. A bounding case was chosen because of this and for the reasons stated in Sections 9.2.2A.6, 9.2.2 intro., and 9.0.2.4. Note the limiting caveat in the first sentence of Section 9.2.2A.6 that we are discussing a "containment or confinement threatening fire". In addition the RMC-line has the largest quantity of suspendable plutonium compounds to be found in the facility in close association with the highest fire loading. Thus it was chosen as the worst or bounding case a technique encouraged in DOE 5480.23.

17. **Please provide the source for the probabilities used (Section 9.2.2B.3).**

Response. The sources are as stated in the text in section 9.2.2B.3. The source for "Fire Present" is based on a reasoned application of Hanford experience. The rooms

on the north side of the second floor are offices, light duty shops, a store room/plastic shop, a lunch room and a large equipment room with very low fire loading. With the caveat that the fire had to be large enough to challenge the sprinkler system we looked at the probability that any given room used by Hanford contractors would have such a fire in a given year. With the thousands of such rooms in use it was clear that such a fire was much less than 1% likely in any year. Had the results of the fire been more radiologically significant that statistic would have had to be refined. The other probabilities are conservative estimates and would always have a heavy component of judgement even if analyzed in more detail. Westinghouse feels these values and references are adequate for the roof fire analysis.

18. **Section 9.2.3.4 should update the source of probabilities to consider incidents past those through 1978 and should compare PFP and Hanford-specific data with the overall DOE data to assess the applicability of and impact of using the DOE overall data.**

Response. The reference cited, Stratton 1967, was inappropriate. It should have been Stratton 1989. He lists the criticalities through the last one in 1978 as stated in the text. There is only one datum for Hanford, the 1962 criticality at PFP. Since there have only been 6 criticalities in the chemical processing industry it is hard to state that our one represents a better or worse statistic.

19. **Sections 9.2.3.4.1, .2, .3, and .4 regarding metal, powder, liquid, and petroleum transport probabilities should provide a basis for the probabilities to complete the discussion.**

Response. The approach to this section is outlined in general on page 9-347. Rather than define a "worst case" criticality in order to bound all others we chose to use the NRC criticality which has radiological consequences that are just significant from a dose perspective. As with fire, there are so many different scenarios possible for creating a criticality that setting a bound on them would be very difficult and would lend little useful knowledge. Rather than define a great number of events, each of which would not meet the criteria of being an accident, Westinghouse chose to show that the combined probability and consequence of criticality were within acceptable limits.

20. **Please provide a comparison to estimated probabilities of 0.45 or 0.23 for the limits to be exceeded as opposed to the 0.025 estimation based on DOE sites history from Section 9.2.3.4 (Section 9.2.3.6.3).**

Response. That was done by the reference to Section 9.4. The reader is given the choice of interpolation from the tables or scaling from the graph.

21. **Please provide reference for these "qualifying" analyses (Section 9.2.4.0.1). From the statement, "Analyses performed for the section have determined that most facilities and much of the equipment will withstand seismic loads," it is not clear whether or not these analyses are credited in the FSAR's seismic analysis.**

Response. As discussed in the response to General Safety Review Question 5 Westinghouse is conducting an extensive review of the systems, structures and components important to safety as identified in the FSAR. A significant portion of the review is the systematic identification of structures used in the safety analysis, review of qualification of these structures, and establishment of a surveillance program to assure the functionality of the structures is not impaired. The effort considers non-seismic forces such as wind loads and dead loads from snow or volcanic ash deposits. This effort will provide the level of clarification desired.

22. **Please provide information to justify the 5.9 g plutonium release for interim risk justification (Section 9.2.4.0.2). The current basis appears weak.**

Response. This section is a summary in an introduction and as such does not contain detailed information. Please see the detailed calculations in Section 9.2.4A.6.3.10, Page 9-488. This review item appears to be almost the same as number 7 under Seismic. Please see further explanation of the basis there.

23. **Please provide the source for the analyses of glove box stability when credit is taken for their remaining upright in a seismic event (Section 9.2.4A.2.12).**

Response. Please see Ocoma (1990b). This 1533 page document examines 59 gloveboxes and 28 filter boxes for seismic qualification. This document has been made available to the PNL reviewer interested in this subject.

24. **The probabilities of Section 9.2.4A.5 appear to be estimated rather than calculated. Why aren't calculational methods used to determine these probabilities?**

Response. The referenced sub-section could be more clear as to the fact that the event probability is a combination of seismic event probability and glovebox failure probability. Methods exist to determine the glovebox failure probability but that these methods (fragility curves) are, at best, extremely uncertain, difficult to defend, and would require extensive structural analysis and possibly some testing. The sub-section could better describe why 10^{-4} was chosen and the effects of the uncertainties. Westinghouse believes that the approach used provides sufficient conservatism to bound the risk associated with this event.

25. **Sections 9.2.1A.2 and 9.2.1D.1.3 should base the probabilities for operator action errors contributing to failures on published human reliability analysis techniques such as Swain and Guttman, NUREG/CR-1278.**

Response. Site specific data is not maintained at Hanford. Therefore, failure rates based on current plant experience cannot be calculated. This forces the analyst to use generic data. The generic data used was Dexter (1980) and NRC (1975). The comment suggests using more current data. There are more current data bases, however, based on our needs they do not significantly differ from the data bases used. Most data bases begin with the WASH-1400 data and claim to add more recent data to it using Bayes or some other method. The basic failure rates are very close to the WASH-1400 data. The addition does not seem to do much. When using Bayes, there is the opportunity to Bayes the same basic data multiple times and bias the data. WASH-1400 was created to analyze commercial nuclear power plants. Many times in the analysis of process plants, WASH-1400 does not have data available to use on the non reactor components. Dexter has turned out to be an excellent source for such data. Dexter is based on process plants at Savannah River where the process contains irradiated materials. This is not normally true of newer data bases based on the chemical process business.

In general throughout the analysis of the PFP facility, a operator failure probability of $1.0E-02$ was used for errors of omission, $1.0E-03$ for errors of commission, and $1.0E-03$ for other errors. Errors of omission imply that the operator is aware of a procedure or out of tolerance condition, but does nothing about it. Errors of commission imply that an operator takes improper action and as a result causes an error. In such calculations, there is no time period considered. If the general philosophy defined in Human Reliability Analysis by Dougherty and Fragola was followed, there would be a time interval in which the action needs to be taken. Generally in process plants there is a significant amount of time available. According to Dougherty probabilities smaller than $1.0E-04$ would be used in these cases. The $1.0E-02$ used in the PFP analysis would be a very conservative screening value and when the overall accident sequence still remains to be less than $1.0E-06$ per year, no further analysis is required beyond screening.

26. **Sections 9.2.4A.2.12.C and 9.2.4A.212.F (reference Ralston 1989) have taken credit for a loss of electric power to the facility at a 99% likelihood for a large earthquake and a 50% likelihood for a small earthquake and have concluded that seismically-induced fires would be limited localized burns that would not cause a credible release hazard. Similar credit is taken in Section 9.2.4A.6.1 for a loss of power to the heating ventilation and air conditioning system in an earthquake and in Section 9.2.4B.1.5, (Table 9.2.4-9) for reducing the probability of a seismically-induced explosion in Glove Box MT-5. This approach is contrary to the worst case modeling approach described earlier. This analysis has not considered the potential impact of multiple fires for seismic events which do not**

result in a loss of off-site power. Crediting loss of off-site power seems nonconservative and doubtful at the stated probabilities, which are based on judgement alone. Please provide a further discussion of these issues.

Response. In Sections 9.2.4A.2.12.C and 9.2.4A.2.12.F the use of 99% and 50% as the failure rate of the electrical system is just part of a qualitative argument. In addressing fire in the PFP the thrust is not to describe all fires but to show that fires will not result in unacceptable releases of plutonium. The primary argument is that the fire loading in the building is low. Reason shows that the probability that a heat source, an ignitable substance, fuel for a sustained fire, and a source of suspendable plutonium would all be present at the same time or location is low. How low is hard to quantify but a probability of 10^{-2} does not seem incredible. In Section 9.2.4A.6.1 the failure of the electrical system is again used only qualitatively.

In Section 9.2.4B.1.5 the failure is used quantitatively but only considers the local building transformers. There are a number of fragile links in the distribution system. None of the Hanford systems are seismically qualified and perhaps not even the regional system to 0.2 g. For example, the large wide transformers at the 251-W Substation stand on narrow bases with no bracing. A structural analysis can show that an item can withstand a particular seismic force but to determine at what probability an unqualified piece of equipment would fail is quite difficult if not impossible. Thus we were left with the professional opinion of a qualified seismic engineer and the conservatism of only considering the fragility of the system at PFP.

The fact that we deviated from the stated worst case approach used in the majority of the analysis is acknowledged and recognized in the document.

Westinghouse has done additional evaluations of the failure probability for the electrical system up to and including the dissolver temperature controller. This more complete, revised analysis provided by Evan Weiner, Internal Memo SMA:EOW:92-202, Nov. 11, 1992, "Estimates for MT-5 Hotplate Power in Design Basis and Small Earthquakes," was provided to the PNL reviewers. The revised figures for power remaining ON indefinitely to the dissolver heaters in the large and small earthquakes are 0.025 and 0.020 respectively.

27. The definitions, values, and references should be included for the following basic events:

ED3	EE1	EG2	EH1	EI1	EF6	EB1
ED14	EE2	EG4	EH2	EI2	EF12	
ED15	EE3	EG5	EH3	EI3	EF14	
ED16	EE4	EG6	EH6	EI4	EF15	
ED17	EE5	EG7	EH8		EF16	
ED7	EE6	EG8	EH9		EF17	
	EE10	EG9	EH10		EF19	
	EE15	EG10			EF20	
	EE16	EG11			EF21	
	EE18	EG12			EF22	
	EE20	EG13			EF23	
	EE21	EG18			EF24	
	EE22	EG19			EF25	
		EG20			EF7	
		EG21			EF8	
		EG22			EF9	
		EG23			EF10	
		EG24			EF11	
		EG25				
		EG26				
		EG30				
		EG3				

Response. These elements were not specifically defined because they either corresponded to the Filtrate Evaporator fault tree (Section 9.2.1B) or to the MT-5 fault tree. The Filtrate Evaporator fault tree was not quantified, as discussed in our response to "Probabilities of Accident Events," comment 1. The MT-5 fault tree was quantified in the Harker (1990) reference. See our responses to "Probabilities of Accident Events," comments 11 and 37.

This and the next two comments were discussed with the PNL reveiwer when he visited the PFP on November 9th. Westinghouse personnel explained the operation of the filtrate evaporator and the product concentrator so that he could better appreciate the meanings for the values assigned to the various events in the product concentrator and for those undefined events shown in the supernate evaporator tree. We also explained our event identification codes which clarified concerns he had in this regard.

28. The values and references should be identified for the following basic events:

ED2	EE7	EF12	EC1
ED4	EE11	EF16	
ED8		EF22	
		EF25	

Response. These elements were not specifically defined because they either corresponded to the Filtrate Evaporator fault tree (Section 9.2.1B) or to the MT-5 fault tree. The Filtrate Evaporator fault tree was not quantified, as discussed in our response to "Probabilities of Accident Events," comment 1. The MT-5 fault tree was quantified in the Harker (1990) reference. See our responses to "Probabilities of Accident Events," comments 11, 27 and 37.

29. The references should be defined for the following basic events:

EE8 EF3 C4 E1
C6

Response. These elements were not specifically defined because they either corresponded to the Filtrate Evaporator fault tree (Section 9.2.1B) or to the MT-5 fault tree. The Filtrate Evaporator fault tree was not quantified, as discussed in our response to "Probabilities of Accident Events," comment 1. The MT-5 fault tree was quantified in the Harker (1990) reference. See our responses to "Probabilities of Accident Events," comments 11, 27 and 37.

30. On Page 9-208, Section 9.2.1A.2, the event EF8 was listed; however, this event was not found in any logic trees. Has this event been used in any of these analyses? If so, where?

Response. The events labeled EF7 and EF8 on page 9-208 are mislabeled and mispositioned. The actual labels for these events are EE7 and EE8 which are shown on page 9-201 on the right hand side of the fault tree branch.

31. The important cut-sets in Section 9.2.1C.3 contain event E-E9. However, this event has no definition, value, reference and/or does not appear in any logic trees. Please discuss the use of this event.

Response. The logic trees were brought over from Harker (1990) incompletely. Element E-E9 appears in the original text. This deficiency will be corrected at the first update.

32. The basic events included in an AND gate "No Correction of Holdup of Solution TK-43" should be rearranged. For example, the basic event EF4 "failure of WF transmitter" is included in the AND gate "No correction action taken by operators." This does not appear to be correct.

Response. The fault tree on page 9-200 is correct. Under the box "No Correction of Holdup of Solution in TK-43" is an AND gate. Assume event EE1 occurs. If the WF fails (EF4) the operators do not have the necessary information to fix the condition of "No Correction..." The AND gate is satisfied. If the WF works then the condition is

satisfied if the operators fail to notice the reading on the operable WF which serves to alert them of solution holdup. Therefore the tree is properly structured.

33. **One of the basic events in the cut-sets list on Page 203 Section 9.2.1A.1, is EH7. However, this event is not defined in the facility safety analysis report. The definition and value should be stated.**

Response. The event should be EE7.

34. **There are two EF7s listed on Page 9-208 Section 9.2.1A.2. Note that these basic events have two different values and definitions.**

Response. The second one should be EE7.

35. **Computer code IRRAS, Version 2.0, has some program errors that produce errors in the cut-set output and the associated calculated probabilities. The cut-sets described for at least one sequence appear to contain erroneous information. Have the results of the IRRAS input/output been validated and verified and/or have the resultant calculations been hand checked for accuracy?**

Comment 38 states, "Please expand the discussion in Section 9.2.1C.3 to include the potential errors in the IRRAS Version 2.0 computer code, the cut-set errors, success criteria, failure definition, individual component failure probabilities, and component mission times. This information is needed to check the validity of the resultant probability of $3.3E-7$ for this event. Also, no section 9.2.4.6 can be found."

Response. The calculations done by IRRAS were also done by hand to verify the results. The probability, $3.3E-7$ quoted above is the 95 percentile. The mean is $1.0E-07$. With respect to the computer program, success criteria has nothing to do with the program. The failure definition is input to the code by the analyst. The individual component failure probabilities are based on a $(\lambda)(\tau)$ approximation or an exponential. For small λ and τ , the multiple approximation is not in error. Mission times are also input by the analyst.

Conversation with Mr. K.D. Russell of EG&G Idaho (DSI 15530-92-013) indicates that no error records are available for version 2.0 of IRRAS. He does not remember any errors related to fault tree quantification or uncertainty calculations.

Westinghouse performed a test of IRRAS and verified all output for cutset generation, fault tree quantification and uncertainties calculations. The only problem identified was a statistically significant error in the reported mean of log-normal distributions. The mean was not used in any MT-5 related calculations. Westinghouse plans to document the test in a supporting document.

36. **On Page 9-318, Figure 9.2.2A-5, "D7" should be a basic event instead of an AND gate.**

Response. This observation is correct and will be corrected in the next general update since the error has no bearing on the conclusions drawn in the FSAR.

37. **Section 9.2.1C.3 should provide an explanation or discussion for the important cut-sets.**

Response. The reference, Harker (1990), contains a more complete discussion of the event and has been provided to PNL in response to their interest in getting a better understanding of the details of this section. Some of the fault tree and cut set information differ between the reference and that found in the FSAR. These have been discussed with Pacific Northwest Laboratory personnel, however, the differences do not affect the conclusions of the analysis.

38. **Please expand the discussion in Section 9.2.1C.3 to include the potential errors in the IRRAS Version 2.0 computer code, the cut-set errors, success criteria, failure definition, individual component failure probabilities, and component mission times. This information is needed to check the validity of the resultant probability of $3.3\text{E-}7$ for this event. Also, no section 9.2.4.6 can be found.**

Response. Please see discussion under comment #35.

A.4 RADIOLOGICAL DOSE

1. **The earthquake analysis assumes that the PFP plutonium (and combustible) inventory are at normal levels instead of the maximum inventory allowed. How can this assumption be justified since individual operations can operate at the maximum allowable plutonium inventories for operating efficiency? Also, operating at values above normal operating limits will result in a nonanalyzed condition for seismic events. Please discuss this issue further and state how above normal levels of plutonium and combustible material will be monitored and controlled.**

Response. The normal levels assumed in the analyses take into account the collective effects of existing proceduralized controls embodied in procedures, specifications and other administrative requirements. Additionally, undocumented practices and expectations regarding glovebox cleanliness, adherence to minimizing personnel doses by maintaining small glovebox inventories and the like also contribute to maintaining the inventories at "normal" levels. Nonetheless WHC has re-examined the analysis to assess the significance of consequences for the earthquake scenario if maximum inventories allowed by criticality prevention specifications were assumed.

The approach used to assess the need for added inventory controls to control potentially available source terms to within those analyzed in the PFP FSAR is described below.

The releases from the facility for design basis and process accidents were subdivided into 5 source areas which cover the entire facility. In order of decreasing magnitude of analyzed releases these facility source areas are:

- o RMC/RMA Process Area
- o Plutonium Reclamation Facility (236-Z)
- o Plutonium Process Support Laboratory
- o Analytical and Standards Laboratories
- o All other areas

The sensitivity of releases and potential for additional source term for each area is discussed below for each source area.

1. **RMC/RMA Process Area** - The RMC process area potentially contains large amounts of plutonium in powder form which is highly dispersable. The FSAR analysts recognized the need for bounding the inventory available for dispersion and the FSAR contains an OSR to administratively control the inventory in the RMC Line to that analyzed which is less the criticality prevention specification limits. The RMA Line is inactive and should have been removed from the facility decades ago. There is no potential to increase the inventory in the RMA gloveboxes. There is a large storage glovebox in this area that currently is physically nearly full of materials destined to be processed in the upcoming stabilization run. The current inventory is well below the FSAR analyzed quantity which is 60% of the criticality specification limit. Filling the few available spaces to the limits would not exceed analyzed values. Two liquid handling glovebox accident scenarios were based on concentrations nearly identical to the criticality limits and about double the normal operational flowsheet values.
2. **Plutonium Reclamation Facility** - The PRF contains most of the inventory in liquid form. Other than at the end of the process where the product is concentrated solution concentrations are routinely controlled to flowsheet values in the tens of grams of plutonium per liter range. A 450 gram plutonium per liter criticality limit is almost universally applied throughout the facility. It is unusual that product concentrations exceed 300 grams per liter. The FSAR analyzed quantities selected are conservative and are above the way the facility is actually operated. It is noteworthy that the Slag and Crucible Dissolver will not be used and about half of the total potential release from the PRF is driven by a postulated explosion in this dissolver. Without its operation this potential

release fraction probability drops to zero although it is currently included in the FSAR accident consequences.

3. **Plutonium Process Support Laboratory** - All gloveboxes which have a potential for significant release were analyzed in the FSAR at criticality limits with two exceptions. These two gloveboxes were assumed to contain 1/3 the criticality limit. If one assumed these gloveboxes were at the limits then the potential increase from the entire laboratory would not increase by more than 50%.
4. **Analytical and Standards Laboratories** - These releases in the FSAR are based on inventories at the criticality limits for most of the gloveboxes. A notable exception is a sample storage glovebox which was analyzed at maximized normal values. Including this glovebox at the criticality limits increases the potential release from these laboratory areas by not more than 50% of what was analyzed in the FSAR.
5. **All other areas** - In summary these areas contain some miscellaneous use gloveboxes, residual quantities held in inactive gloveboxes and filtration equipment. Individually each are not big inventory areas and the FSAR basis is not particularly sensitive to changes in their combined inventory relative to the above four source areas.

The supporting calculations for the FSAR were reexamined and doses were re-evaluated using individual glovebox and process area inventories at criticality prevention specification limits rather than maximized normal inventories. The above differences in potential releases from the PFP and the resulting doses were summed. The following summarizes the calculations:

- o For the 10^{-4} per year probability earthquake scenario the potential doses increase by 34% over those contained in the FSAR.
- o For the 10^{-6} per year probability earthquake scenario the potential doses increase by 14% over those contained in the FSAR.
- o The differences in the ratios between the two different earthquake scenarios is attributable to the details of scenario development and the degree of conservatism used by the analysts who prepared the FSAR.
- o These higher potential doses are still well with the risk acceptance guidelines and in retrospect should have been the values presented in the FSAR to eliminate concerns on the need to provide additional controls on plutonium inventories in the facility.

2. **The GENII code has built-in building wake equations. Why weren't they used? The necessary building cross-sectional areas are defined in Chapter 1 of the SAR. Apparently the building wake factors derived from the Nuclear Regulatory Commission's Regulatory Guide 1.145 have only been applied to ground-level releases; however, the text in the SAR implied that they had been improperly applied to elevated releases also. The text should be changed to reflect how the building wake factors were actually applied.**

Response. The GENII code has only a building wake correction. The USNRC Regulatory Guide 1.145 correction accounts for both building wake and plume meander (due to changes in wind direction during the release). In this particular case the plume meander effect dominates so that the correction factor is independent of building cross-section. It is agreed that the FSAR text should be edited to clarify that this adjustment is applied only to ground-level releases. A proposed change to Page 9-592 has been prepared for inclusion in the next general update of the FSAR. This clarification does not impact conclusions drawn in the FSAR.

3. **Why, in Tables 9.3-4, is the height of release from the stack given as 61 m for acute releases rather than the 89 m used for chronic releases? Is the ventilation flow significantly reduced during accidents?**

Response. The amount of plume rise and, therefore, the effective stack height is a sensitive function of local meteorology around the stack, and to stack exhaust velocity and temperature at the time of the accident. The local meteorology is not known for a postulated accident, and large uncertainties exist in the stack exhaust conditions, so the physical stack height is normally assumed for conservatism. On an annual average basis, however, normal stack exhaust conditions can be assumed, and variations in local meteorology average out well enough to assume an effective stack height which is used for chronic releases only.

4. **The wording in the Van Fleet reference indicated that the dose calculations for ingestion and external exposure properly included the entire 16 g of plutonium and not just the 3 g of respirable size used for inhalation. This important point should be included in the text because it explains why the dose from ingestion is such a large fraction of the total dose.**

Response. This discussion is already included in the text (see Page 9-579, second paragraph from bottom of page). Additional discussion with the PNL reviewer indicates that the problem lies with the last sentence of the Inhalation paragraph. In a future revision for clarification we will add words to indicate that the conservatism in use of 1 μ m particle size in GENII is in the over-all scenario dose since the material released from the building is considered to be respirable from 10 μ m on down.

5. **The assumption of Class Y plutonium for the oxide release is proper, however, shouldn't the dose calculations also account for the conversion of plutonium to a more soluble form ($f_1 = 1 \text{ E-03}$) during its incorporation into foods, as recommended in International Commission on Radiological Protection Publication 48? Page 9-591 indicates that Class Y was also used for ingestion. Is this how it is used?**

Response. Westinghouse Hanford Company has been directed by DOE to conform to ICRP 26 and 30, which the GENII code, along with the user manual, does. Westinghouse Hanford Company has not received any direction from DOE relative to ICRP 48. Even so, this point has been under investigation within the company for some time, but has not yet been resolved. It is felt that nearly all of the plutonium ingestion dose from an accidental release, if it occurred, would be by direct ingestion of particulate material stuck to the outside surfaces of harvested crops. It is expected that this material would still be in an oxide (insoluble) form. The subsequent ingestion of material incorporated into plants through root uptake (presumably after conversion to a more soluble chemical form) accounts for only a very small part of the total ingestion dose for plutonium. New requirements for risk determination are contained in the latest revision of the Westinghouse Nonreactor Safety Analysis Manual, WHC-CM-4-46, which provides for calculating the dose to the offsite receptor without regard to ingestion for the purpose of risk determination. Thus, as soon as the annual revision to the FSAR is made the question of what ingestion model to use will become less critical. It will not go away entirely as we will still be publishing the combined dose for information purposes. Calculation of dose involving ingestion requires numerous modeling choices that make it a poor candidate for use in risk acceptance. Choices include crop interdiction, season of year and duration of ingestion. In addition it is not always clear as to the mix of class W and Y material released. A Hanford wide committee has been established to try to come to some unified solution.

6. **The reference to Till (1983) in the footnote to Table 9.3-5 is apparently carried over from another reference because it is not given in the reference list. In any case, the reference for internal dose factors recommended by DOE is DOE/EH-0071 (1988), "Internal Dose Conversion Factors for Calculation of Dose to the Public." Please comment on this further.**

Response. The footnote to Table 9.3-5 is completely erroneous and should be deleted. This will be included in the next FSAR update since it doesn't change any conclusions pertaining to the risk of operating PFP.

7. **Because this SAR has been compiled from several older ones, I noted that Table 9.3-6 was called out as Table 9.3-7 on Page 9-601. Only one of these two tables defined the Translocation Class of the released plutonium and this error has made it difficult to determine what class was assumed for the other. I have**

concluded that the first of the two tables was referring to Class Y plutonium. Is this correct?

Response. Yes. Table 9.3-6 is based on the analysis by D.C. Deere who assumed class Y solubility. This should have been included in the table title. The information in Table 9.3-7 superseded Table 9.3-6 and Table 9.3-6 could be deleted altogether. These corrections will be included in the next general update of the FSAR since they have no bearing on the conclusions drawn in the document.

- 8. The Glossary gives an obsolete definition for the term "Permissible Dose," this term is no longer recommended for use.**

Response. Westinghouse agrees, limiting dose criteria are now given in terms of the Radiological Risk Acceptance Guidelines described in sections 9.4.1 and 9.4.2. The term "permissible dose" should be deleted from the glossary. We welcome any comments regarding the glossary. Often subject matter specialists in the many subjects addressed are the only ones who recognize out of date definitions. The glossary will be reviewed and Westinghouse will delete the definition at the next FSAR update.

- 9. The glossary uses an obsolete value for the number of ergs produced in a gram of air by one roentgen of X or gamma radiation. The correct value should be 86.9 in Handbook 85 of the National Bureau of Standards.**

Response. Westinghouse agrees. The value 86.9 ergs/g of air is also given in The Health Physics and Radiological Health Handbook (1984). Westinghouse will correct the definition at the next FSAR update.

- 10. Someone could argue that the site boundary should be the locations along Highway 240 and the Columbia River for the highest E/Q values. These locations are accessible to the public and a plume from an accident of a few hours duration could pass over such areas before anything could be done to remove the public or prevent their entry. Does Westinghouse Hanford Company have an emergency plan to remove the public and control these locations during an accident? Please discuss this issue further.**

Response. For the Hanford Site, the site boundary forms the limit of the "exclusion zone" as defined in 10 CFR 100.3. According to the definition the exclusion zone "may be traversed by a highway, railroad, or waterway, ... provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety." Extensive emergency plans are in place to control and evacuate onsite members of the public in the event of an accident. Risk acceptance criteria for members of the public in an accident are ultimately based on the NRC 25 rem siting criterion at the exclusion area boundary (i.e. the site boundary), and do not apply to members of the public on

highways, railroads, or waterways within the site boundary. The new revision to WHC-CM-4-46 will define receptors to be either onsite (including members of the public within the site boundary) or offsite (the general public).

11. **Page 9-592, last paragraph, states that the cross-sectional area used for the building wake factor calculation was 600 m². The Van Fleet reference states that 100 m² was used for the cross-sectional area. Why is there a difference and why is the calculated wake factor the same in both instances?**

Response. Westinghouse has carefully reviewed the calculations performed by Van Fleet and cannot find a 100 m² building cross-section in Van Vleet's analysis. The correct value is 600 m² as given in reference 3 of Van Vleet's document. Even at 600 m² however, the minimum building cross section is too small to have an effect and the plume meander dominates (see item 2).

Westinghouse will clarify this point in the next FSAR update.

12. **Why is the plutonium assumed to have aged only 5 years in one dose calculation (Page 9-588, Table 9.3-5) and 35 years in another (Page 9-591, Table 9.3-8)? The average age of the plutonium in the PFP will probably be somewhat greater than 5 years by the time operations resume. Since the SAR states that the dose impact from plutonium increases with age to a maximum at about 40 years, an age greater than 5 years would seem to appropriate. Please discuss this further.**

Response. Van Vleet's analysis gives results for the Pu mix aged both 5 years and 35 years to permit interpolation to an intermediate age. The results show, however, that the differences between the doses for the two ages are insignificant (5% to 6% increase from 5 y to 35 y).

The 35 year material is assumed to be located in the duct work of the old incinerator building, 232-Z. When the building was placed in service in the early 1960s there was a backlog of feed material from the mid- to late-1950s, thus the figure of 35 years was used in 1990 when the FSAR was written.

The age of 5 years was based on the date from americium separation which will take place in the CA column of the PRF or from the Pu cycle in PUREX. Not long into the up-coming campaign, the majority of the plutonium in the PRF will be much less than 5 years and will be lower in Pu-241 than what is shown in the FSAR. Use of the mixture as shown is conservative since the Pu-241 makes up a significant portion of the total activity in the mixture as shown. With a $t_{1/2}$ of 13 y for Pu-241 the material long out of the reactors but short out of chemical separation will have significantly less activity than that shown. There is no way to accurately characterize all the isotopic mixes that might be found throughout the plant but the mixture shown was selected to be representative of that material available for release.

Westinghouse will update this discussion in the next FSAR update.

13. The footnote to Table 9.3-5, Page 9-588, states that doses from a somewhat different isotopic composition would be no more than 50% higher than that calculated for the mixture given in the table. However, there was some 12% plutonium produced in 100-N fuel and that mixture contained a much greater (weight) percent of Americium-241. It appears that the dose from that mixture may be noticeably higher than for 6% plutonium. A composition for 12% fuel derived by Rockwell Hanford personnel is given in Table 3.2, Page 3.12 in DOE/EIS-0115D, dated April 1986, Draft Environmental Impact Statement, Process Facility Modification Project. Please discuss this further.

Response. Westinghouse agrees that the footnote to Table 9.3-5, Page 9-588, is incorrect (see item 6). The doses on a per weight basis are driven mainly by activity (not dose conversion factor) of the various isotopes. The operative parameter here is the product of the specific activity (Ci/g) and the dose conversion factor. The implications of processing 12% Pu 240 material is included in Section 8.2.1.5, "Characteristics of Contained Sources in the PFP" which addresses plutonium of even higher Pu 240 content.

14. The statement in the second paragraph on Page 9-599 is incorrect. I can find no evidence that there is a time of 5 years specified for a "continuous" exposure. In any event, I can find no clear reference to DOE guidance concerning the choices of the dose versus probability values used in the SAR and illustrated in Figure 9.4-1 on Page 9-597. Please discuss this further.

Response. Westinghouse agrees that the statement is incorrect. DOE Order 5400.5 references on page 9-599 are apparently out of date. Figures 9.4-1 and 9.4-2, the radiological and toxicological risk acceptance guidelines, are derived from WHC-CM-4-46.

At the next FSAR update that section 9.4.2.1 ("justification" for the guidelines) will be deleted, and a reference WHC-CM-4-46 substituted.

15. Have the concerns about detection capabilities for some nuclides, reported in Schwendiman, L. C., and J. A. Glissmeyer. 1992. Preliminary Evaluation of the Gaseous Effluent Sampling and Monitoring Systems at the 291-Z and the 296-Z-3 Stacks - Plutonium Finishing Facility. PNL-8036, Pacific Northwest Laboratory, Richland, Washington, as having been found in 1976, been corrected?

Response. The document evaluated the system which was replaced about ten years ago with a system located at the fifty foot level of the stack. Deficiencies identified in the document are moot since the evaluated equipment has been abandoned in place.

A.5 FIRE AND EXPLOSIONS

1. **The contractor should consider whether a seismic event can cause a loss of electric power, loss of steam flow, loss of ventilation, loss of fire water, and fire in one or more cells. The Loss of Ventilation analysis in Section 9.1.8 calculates a reverse flow under certain loss of ventilation conditions, but doesn't consider a fire in conjunction with loss of ventilation. Please address whether such fires under loss of ventilation conditions can cause a heat-driven reverse ventilation flow out through the supply air filters to the environment.**

Response. Westinghouse agrees with the first sentence and indeed included such analyses. Several places in the seismic analysis either explain the general approach or present the details. Please refer to Sections: 9.2.4A.2, "Seismic Accident Scenarios", introduction and the following scenarios (page 9-387); 9.2.4A.2.12 "Accident Scenarios-Fire" (page 9-407); and 9.2.4A.5 "Probability" (page 9-415).

Theoretically, a ventilation reversal could take place and would probably take place in a large fire in the second floor rooms discussed in the roof fire scenario (9.2.3B). Examination of the probability elements in the RMC-line fire scenario (9.2.2A.2) (page 9-316), the general discussion on fire in Element D-1 (page 9-40), and the seismic fire scenario referenced above shows that the probability of a fire with the energy to generate a significant backflow is quite low and coupled with a significant release of plutonium even lower. Descriptions of the redundant nature of the ventilation system in Sections 5.4.1 "Ventilation Systems" and 5.4.2 "Electrical" and the introductory paragraph for Section 9.1.8 "Loss of Ventilation" show that complete loss of ventilation, has a low probability. These combined probabilities coupled with the probability of a significant plutonium release are very low and thus did not make the initial screening.

Many of the arguments presented in reply to comments 3 and 4 below apply to a reverse flow situation as well. Westinghouse and PNL have met to discuss this issue and it was explained that the fire loading is low, mechanisms of spread are reduced, there is no positive motive force, and the ducts have a considerable area for heat loss. In addition the arrangement of the supply ducts would tend to drive much of the smoke and gases up to the 2nd floor into room 308 rather than out of the building via postulated reverse flow paths.

2. **Do fires or explosions have the potential for igniting dust explosions in the duct work? If this has not been considered, please discuss this possibility further.**

Response. In about 1987, a study was made of dust contamination in ducts. No records could be located but in early September 1992, the cognizant, ventilation system engineer telephoned the retired man who was in charge of the 1987 inspection. He stated that the ducts were essentially dust free. The reasons for this are that the only

significant source of dust is from the cotton protective clothing and that the E-3 inlet grills are all protected by fiberglass roughing filters that remove significant fractions of that dust that is generated.

3. **The fire accident analyses in Section 9.2 rely heavily on the cooling effect of mixing the hot gases from the fire with cool exhaust flow from rooms uninvolved in the fire to conclude that the HEPA filters will not be exposed to unacceptably hot gases. There are several problems with this approach:**

- **Concurrent loss of ventilation conditions are not considered; this would result in substantially less mixing with cooler air from other rooms.**

Response. Without positive, vigorous air flow a number of factors that increase the severity of a fire and its effect on filters would be absent. Loss of ventilation in the period just previous to the start of the fire would mean the evacuation of the Zone 3 areas of the building so that no maintenance or operating activities would be taking place. An evacuation due to ventilation failure is an orderly, deliberate event. Non-essential personnel simply leave their work place and file out of the radiation zones. Essential personnel remain behind either with mask at hand or in mask depending upon the nature of the ventilation upset. These people perform orderly shutdown of processes or jobs. The Hazardous Work Permit, (Form 54-6700-148(02/89)) requires a fire watch person to remain at the site of cutting/welding for at least 30 after job completion. This person would be considered as one of the essential personnel. This would decrease the probability of an ignition and also reduce the probability of human detection. The failure of human intervention is given a very high probability as written. The rate for failure of the Halon system is based on a back-fitted suppression system that has to act while the ventilation is flowing. As stated in fault tree element C-3, the reason Halon fails to suppress the fire is that air from a burned or melted glove sweeps the Halon away. With a ventilation failure the Halon concentration would be quite high and much more likely to suppress even a deep seated, smoldering fire in ordinary combustibles.

As stated in Section 9.2.2A.1.1, Page 9-307, the motive force for spread of the fire to all parts of the glovebox system is the normal air flow. Without this air flow the fire would be much more limited in location. Subsequent loss of air movement might mean heat build-up in the room of ignition. But that could result in displacement of glovebox windows from their frames so that they would slump and fall or drip to the floor of the glovebox or the room. With the majority of windows being polycarbonate material, simple removal from impingement of flame would cause them to self extinguish. Melting is also an important heat removal mechanism for other types of plastics as they sag or drip onto surfaces with temperatures below the combustion temperature.

With a reduction of combustion air the exhaust gas temperatures would be reduced as in an ordinary wood stove with the supply damper closed. The effect of the long runs, high surface area, and large mass of the duct systems leading to final filters would be significant in comparison to the total heat output and the lower temperatures ((Figures 9.2.2A-2,3 and 4 (9 Sheets))).

Filter plugging would become much more significant without suction in the exhaust plenums. The scenario as presented on Page 9-308 shows failure of the filters due to smoke loading with a resulting pressure differential due to a functioning ventilation system. With the system down this would be less probable.

Without ventilation flow the heat loss to the E-3 room exhaust system would be minimal as the exhaust grills are near floor level and even slight smoke loading of the fiberglass dust-stop filters would effectively restrict flow.

- **The possibility of buoyancy-driven reverse flow in the duct system is not considered.**

Response. While a buoyancy-driven reverse flow is not considered, neither are the various particle removal mechanisms considered quantitatively, some of which would predominate in a slower moving system. Thermophoresis, diffusiophoresis, and electro-static precipitation would all play a larger role. In addition, the initial suspension of plutonium particulates would be significantly less in the lower air flows.

Without an external motive force the proportion of gas going into the room would increase and the probability that the sprinkler system would be effective in heat removal and particle removal would increase. The rather high probability of failure given for the sprinkler system is, as stated in Element C-5, Page 9-322, not due primarily to the failure of water to flow out of the sprinkler heads but to the arrangement of the fuel inside the gloveboxes. Thus a fire in which the primary hot gas flow was into the room would be more likely to be mitigated by the water as a heat and particle removal agent than is shown in element C-5.

The velocity of reverse flow air due to buoyancy in a few rooms of a building of this size would be low. With such a low flow particle removal to the walls of the supply ducts and in the wet and dry pad filters of the air supply system would be more effective than is generally credited with normal air flows.

- **The possibility of several fires being started in different rooms by a seismic event is not considered.**

Response. Multiple, seismically induced fires are considered in Section, 9.2.4A.2.12, Pages 9-407 through 9-414.

Please discuss these scenarios further.

Response. The scenarios presented are not intended to delineate all the possible ways that a fire could effect the facility. As discussed in the introductory paragraphs to Section 9.2.2, these fire scenarios are intended as worst case in terms of radioactive release. As fire is essentially a chaotic process, small changes in initial conditions can lead to large variations in result. No written scenario can hope to duplicate in any but the broadest terms what will actually take place. The scenarios present the reader with a general indication of the probability and consequence of a radiation release due to a fire in the PFP.

4. **Please evaluate whether the wooden floor of the Zone 3 supply plenum can be ignited by hot combustion gases from a fire elsewhere in the facility under loss of ventilation conditions that allow buoyancy-driven reverse flow in the ventilation system. Such a fire in the supply plenum floor might significantly increase the buoyancy-driven reverse flow. Such a flow increase could exacerbate the fire and mobilize radionuclides in the burning room. In addition, the supply plenum fire might destroy the supply side filters, which are the only barrier to free release to the environment in this scenario. Please discuss these scenarios further.**

Response. The floor of the main supply plenum is made of metal roof decking covered with a waterproof seal. The rooms adjacent to the plenum have much lower fire loadings than those on the oxide producing areas of the RMC-line. The supply ducts leading to the RMC-line rooms is approximately 200 ft long. The majority of the duct has a cross section of 6 ft by 4 ft. The effect of heat loss in ducts downstream from fires was studied by Norman Alvares and published in UCLR-52819, August 1979 and in the FSAR reference Alvares, et al, 1980. They indicate that the integrity of filters, in a powered exhaust system is protected if the filters are more than 10 duct diameters from the entry of the heated air that is below about 300°F. Equation 4 in Alvares (1979) shows the outflow temperature of a duct to be reduced as the area of the duct is increased and as the mass flow rate decreases. Both these terms are part of the expression as a negative natural logarithm so the effect is more than linearly expressed. The largest ducts in Alvares' work is a 24 in. in diameter. Thus the supply plenum from the RMC-line to the main plenum would have much greater heat loss characteristics due to the increase area and flow rates below that of a powered system. The other plena in the building are larger. Rooms closer to the main supply plenum will have a shorter run but the rooms do not have anywhere near the installed fire loading. The main supply plenum itself would absorb significant amounts of heat. The section of the L shaped plenum where the Zone 3 supply plena attach is 20 ft wide about 13 ft high and 68 ft long (H-2-16173). (Shown in outline on Figure 1.2-

10). This and the other leg of the plenum would absorb a considerable amount of heat.

Not all buoyancy-driven flow would be in the reverse direction. The S-3 distribution duct has several risers leading to the second floor in the radiation zone. Two 24 in. by 36 in. risers take off at about column line 15 and 12 and an approximately 48 in. by 48 in. riser at about column line 6 (H-2-24607). Since these terminate at about the same level as the inflows for the supply fans but into a fixed space it is hard to predict which direction the flow would travel. In any case these are largely unoccupied spaces. The one occupied space, Room 341, is being used as a hardware stock-room but will be abandoned due to radiological concerns in a few months.

Mobilization of radionuclides would be less from reverse flow than from powered release as described in Section 9.2.4A. The powered flow is assumed to be inside the gloveboxes where the suspend powders exist.

It is unlikely that heated air moving vertically through a few 18 in. diameter room-supply ducts into distribution ducts of the size given above would have a velocity greater than the flow from the powered exhaust system described in Section 9.2.2A.

If the wet- and dry-pad filters in the supply system were destroyed the effect on plutonium release could hardly be greater than the release out the destroyed HEPA filters assumed for the scenario in Section 9.2.2A. Lower flow rates and large cool duct surfaces would remove particles as described in comment 3 above.

5. **Please provide further discussion on the effects of Halon on a plutonium metal fire (Section 9.2.2A.2.1, Item C-3). Identification of the possibility of plutonium metal being present in Halon covered areas would also be appropriate, such as producing plutonium oxide from existing plutonium metal.**

Response. Please read the discussion of plutonium burning in Halon atmospheres presented in ISO-756, R. E. Felt, "Burning and Extinguishing Characteristics of Plutonium Metal Fires", August 1967, Isochem Inc., Richland, Washington. Felt shows that under certain conditions Halon increase the temperature and burning rate of plutonium metal. He also concludes that at the normal concentrations used in extinguishing systems and at even higher concentrations but at normal massive plutonium burning temperatures, Halon has no effect on burning rates and temperatures.

Plutonium metal will no longer be produced on the RMC-line. Occasionally plutonium metal must be removed from cans due to in-leakage of air with resulting chance for pressurization. During repackaging the metal is exposed to a glovebox dry air atmosphere. This work is done in the Repackaging Glovebox in Building 2736-ZB where there is no Halon.

6. **Due to the potential for filter loading from the discharge of dry chemical fire extinguishers, the location(s), use, and administrative controls of these extinguishers should be addressed. Where dry chemicals are used for fire suppression, the potential impact on the filters should be evaluated.**

Response. Filter loading would have no effect on release of plutonium from buildings and like many industrial hazards is not discussed in the FSAR as directed in NRC Guide 3.39.

Gloveboxes have HEPA filters locally at the exhaust exit that could be loaded. The gloveboxes connected through conveyer systems to a common dry air system could suffer local plugging but with 8 lb extinguishers and many filters in remote gloveboxes the effect would be minimal. In the single dry air glovebox in Building 2736-ZB, the automatic controls on air pressure within the Glovebox would control the pressure relative to the room. The dry air systems in Building 234-5Z are connected to four parallel filter boxes each equipped with 2 parallel 24 in. by 24 in. by 11.75 in. HEPA filters and the E-4 filter room has 112 such filters. The E-3 or room exhaust is filtered at the exhaust grills in the rooms by a fiberglass roughing filter and then go to 3 parallel filter rooms each equipped with 112 HEPA filters of the size described above. Other buildings have fewer filters but each has a large number in proportion to the size of our extinguishers.

Room-air supplied gloveboxes can have the exhaust filters plugged but this only reduces the negative in the box and the powder would have to compete with smoke to see which would plug or restrict the filters first.

7. **The analysis for the acetylene explosion pressures (Section 9.2.1F) should include the contribution from the solvent (acetone) within the cylinder.**

Response. The sensitivity of the analysis was examined to determine the effect of acetone on the results of the calculations. In a recent telecon with a Matheson Gas Company representative we were informed that since the acetone is absorbed onto an inert material within the cylinders, only about a quart of excess acetone would be likely to escape. The heat of combustion for acetone is about 65% that of acetylene. This quantity is considered to be negligible in energy contribution compared to the total quantity of acetylene. Therefore, the omission of acetone does not significantly affect the results.

8. **Was the maximum explosion pressure (due to acetylene in a small room) used for all rooms to account for the possibility of gas pocket (local explosions) in larger rooms?**

Response. Section 9.2.1F.3 shows that two cases were considered. A 3000 ft³ room was assumed. Explosion energies were based on large and small bottles. In addition,

page 9-281 last sentence addresses gas pockets and Table 9.2.1F-1 on page 9-285 shows that building damage and plutonium release was considered even if a flammable mixture was not present (failure sequence 1, 2, 3, 4, and 6).

As a result of this FSAR review the calculations involved with the acetylene explosion appear to be in error. Westinghouse is reviewing the calculation methods and the methods to control the volume of acetylene that can be taken into the building. In addition we are investigating the nature of flow limiting valves located within cylinders as are specified in 49 CFR. When new information and controls are developed the entire scenario will be re-written with attention on limited volume explosions as these may well be the only kind possible. They are likely to be on the order of miniature unconfined vapor explosions as are commonly encountered outdoors.

Westinghouse recognizes that an explosive force of 60 lb/ft² when applied over a large section of 8 in. thick concrete wall would cause failure. Since it is impractical to calculate all the possible explosion scenarios in all rooms of the PFP and acetylene is seldom used, there will be a revision concerning the use of acetylene. Acetylene will be banned from use in the parts of the building where an explosion could cause a spread, either on-site or off-site, of radioactive materials or chemicals at significant levels unless case by case analysis indicates an explosion would not damage confinement integrity.

9. Where in the SAR is the evaluation of missile hazards from compressed gas cylinders?

Response. It does not exist. The probability and effect of such an event is quite low. The effect could be to breach gloveboxes which is covered in general in Section 9.1.9 or to disable equipment which is discussed under the individual accident analyzed. While this might bias the failure statistics as presented the magnitude of bias would be slight. The probability that a flying bottle would effect a specific device is even less probable. The PFP has no equipment that is imperative to have in place for safe shutdown that could not be shut down manually in a reasonable amount of time following such a dramatic event. The kind of damage caused by a missile is enveloped by the wide spectrum of damage assumed in the seismic analysis in Section 9.2.4. A bottle could be propelled through a glovebox and out through an exterior wall carrying plutonium contamination with it. The consequence would not be great as can be gauged by the spill scenarios. The building ventilation system would very quickly draw in air through the hole from the outside. This breach in the exterior wall could be quickly patched to keep out adverse weather conditions and then fully patched following decontamination efforts. The cylinder and ground would be easily decontaminated. There are few gloveboxes near exterior walls and most of them are in small rooms where a bottle would be unlikely to attain enough velocity to penetrate both a glovebox and an exterior wall.

10. **An evaluation of the effects of fire on safety Class 1 items should be specifically addressed (direct impact, power interruption, and signal interruption).**

Response. There are very few safety class systems in the PFP as shown in Table 4.4-2. None of them have a function the failure of which could not be coped with manually. For example, if fire effected the programmable controllers for the dissolver pots in MT-5 the power could be shut off in a number of locations if required. If a fire effected the seismic shut off system for the supply fans, the fans could be shut down manually if required. The controllers on the pots in MT-5 are the only safety class 1 devices that specifically prevent an accident in case of human failure. All other items are either static, such as building walls, or are mitigative or detecting. Fire is such an obvious event that mitigating actions would take place.

11. **An evaluation of the effects of fire-water flooding on safety Class 1 items should also be directly addressed, i.e., structures, continuous air monitors, etc.**

Response. Please see the response to Item 10 above.

12. **Documentation of the effects of fire-water flooding on below-ground facilities, e.g., exhaust fans 291-Z, was not found. Please provide a discussion of this issue.**

Response. Internal flood in general is discussed in Section 9.0.2.4, Element A-5 Page 9-30.

13. **0200-99.0.2 of DOE6430.1A requires the consideration of potential hazards from off-site sources that could affect safe operation of special facilities. Possible off-site hazards that should be evaluated for the PFP include:**

- **brush fires (loss of off-site power and steam)**

Response. Loss of power and steam is discussed with the accidents where that kind of loss has an effect and in no case does such a loss lead to an onsite or offsite exposure.

- **smoke fouling of intake air**

Response. The aerial photograph in Figures 9.0.2-5 and 9.0.2-6 give an indication of the distance from the vegetated areas and the PFP. While smoke from range fires can be quite irritating it seldom or ever is truly incapacitating even to fire fighters. With the dilution due to distance, and the action of the dry and wet filters and duct surface removal the effect to facility personnel would be minimal. Alvares et al,(1980) shows that it takes very dense concentrations of smoke to effectively plug filters.

- **toxic chemical release (tank farm, other facilities, truck, and rail car)**

Response. In no case would intake of noxious chemical per se cause a significant release of plutonium. Site wide and facility emergency plans address this problem and procedures are in place to cope with them (Section 10.5).

- **radioactive releases**

Response. See the comment above regarding chemical release.

- **coal plant explosion/missile generation**

Response. The coal plant is about 2/3 mi ENE of Building 234-5Z and generates steam at a pressure of 225 lb/in.². The north and east sides of both Buildings 236-Z and 234-5Z do not contain plutonium. A distance of 2/3 mi is a safe distance for an inhabited building from an explosives magazine containing 250,000 lb of explosives (DOD 4145.26M, "DOD Contractors' Safety Manual for Ammunition , Explosives and Related Dangerous Material" October 1968).

- **refueling tanker fire/explosion**

Response. Refueling tankers outside the PFP yard area are too far away from the facility to be of concern. Within the PFP fenced yard vehicles move much more slowly than highway speeds and traffic is sparse at best. Transportation accidents within the yard were dismissed as inconsequential by the analysts and are not documented. In a similar fashion the analysts assessed and dismissed the risk associated with refueling tanker accidents during refueling. These operations are infrequent and not conducted adjacent to plutonium processing or storage buildings.

- **temporary diesel generator fire/explosion**

Response. Temporary diesel generator installations outside the PFP are too far away to be of concern. A temporary diesel installation was established near the south fence line of the PFP yard away from the main processing and storage buildings. Use of this diesel has been discontinued. Nonetheless its location was distant enough not to be of concern with respect to a plutonium release from PFP. Environmental considerations require that above ground diesel tanks be equipped with a berm or containment basin large enough to hold the entire contents of the tank. This prevents spread of fire by flow of burning fuel as well as providing environmental protection.

14. **The SAR should be updated to include the progress of Project B-631 PFP Fire Safety and Loss Limitation, to bring 234-5Z to the National Fire Protection Association code.**

Response. Project B-631 was scoped to provide additional controls to limit property loss and enhance employee life safety from a fire. Completion of this work will not change the conclusions of the FSAR.

15. **What are the potential consequences of a duct-level explosion and fire [Potential source: battery room and transient fire loads (filters, wood, plastics, cardboard, etc.)]? Could the exhaust ducts be damaged such that the reduced room air pressure cause the metal floor to buckle? Please provide further discussion of this accident scenario.**

Response. A duct level explosion would in all cases be less than those used to envelope these accidents in Sections 9.2.1 and 9.2.2.

The battery room is in the Zone 1 part of the duct level separated from the Zone 3 duct rooms by an 8 in. thick reinforced concrete wall. There are no sources of plutonium in Zone 1. Zone 3 contains only a few kilograms of material inside stainless steel ducts separated in some places by 2 concrete walls.

Transient loads can be in any location but the rooms in the duct level are very big and almost entirely devoid of installed combustibles. Thus, although the fire loading per square foot could be high in a small area many of the factors that magnify the effect of a fire would not be present in the large rooms. Heat would not be trapped to dry and pre-heat the fuel before impingement by flame and to cause general structural damage. Gas generation in fuel separated from flame and collection to flash-over would be highly unlikely. Reflection of radiant energy would be reduced. Color coded, self extinguishing plastic is used for temporary enclosures. Scaffold planks are difficult to ignite in the absence of a intense external heat source and become insulated by a char layer. The wood frames for HEPA filters are fire retardant treated.

Rooms in the duct level are equipped with fire suppression wet pipe sprinklers.

Upward buckling of the duct level floor would not in itself lead to a release of plutonium. The likelihood of this is slight as the duct level floor is composed of roof decking that is unsealed so air can come through over the entire expanse of the floor. The pressure in the exhaust plenum, down stream of the filter rooms is -6 in. water (Figure 9.1.7A-4) so this is the least pressure that could be obtained in the duct level assuming that it is well sealed. The filter rooms would not be effected by such an occurrence as they rest on the concrete second floor above the duct level. This is about 31 lb/ft² which is not a great pressure for expansive building areas such as floors, roofs and walls.

16. **What controls or systems are in place to prevent reactive chemicals being introduced to the 241-Z facility?**

Response. The Pu Operations procedures Z0-100 and 101 series provide detailed operating instructions to personnel on chemical make up operations the 241-Z facility. Chemicals used to decon or perform maintenance or operational activities not covered by the chemical make up procedures are addressed in the Administrative Manual WHC-CM-5-8, 3.13, Written Hazard Communication Program at PFP. In addition to the chemical safety information provided by 3.13, a new procedure, 5-8, 7.11 (new section) Hazardous Material Management Plan has been written to address the requirements of WHC-CM-4-3, C-7 Chemical Storage and Handling, as well as all applicable OSHA and NFPA requirements relating to safe storage and handling of hazardous materials and chemical products. Finally, only authorized personnel are allowed entry into the key controlled 241-Z facility and authorization is only maintained by the individual's completion of the 29 CFR 1910.120 Hazardous Materials Operation and Emergency Response training course.

17. **What effects will fires have on chemical and product contents, e.g., the production of red oil, within the affected rooms?**

Response. The primary effect would be for liquid chemicals to boil away with some degree of decomposition and for some dry chemicals to decompose non-violently. A list of process chemicals is given in Figure 9.2.4-7 and a description of chemicals in the PRF in Section 9.2.4A.6.3.2.A.o on Page 9-469. The chemical storage areas and chemical make-up rooms have very low combustible loadings.

"Red Oil" is the name given to a compound made by heating a mixture of tri-butyl phosphate, and a heavy metal such as uranium or plutonium in the presence of high molar nitric acid. This mixture is not present in the chemical make-up or storage areas. Such a compound could be made if there was a large and sustained fire in the column portion of the process cell but there are no combustible chemicals in the cell. The tri-butyl phosphate is made up as a 20% by weight solution in 80% carbon tetrachloride, a formerly common fire extinguishing agent. Thus a fire that caused a leak of solution into the access glovebox would be extinguished.

18. **What limits are placed on the chemical concentrations and mixing of potentially explosive compounds? For example, is there an operational safety requirement limit requiring that all shipments and processes use less than 30 wt% H_2O_2 ? Since concentrated nitric acid has the potential to produce explosive compounds with organics in addition to tributyl phosphate (glycerin, alcohol, etc.), what controls or concentration limits are placed on the feed stock, operations and especially waste disposal to preclude the generation of explosive mixtures or compounds?**

Response. All waste is handled in accordance with WHC-CM-5-8, 1.30 Solid waste Management at PFP. In addition, the newly written Hazardous Material Management Plan, (see above), and WHC-CM-5-8, 6.6 Bill of Material Processing requires that the Hazardous Material Specialist review and approve all chemical material orders. This review includes an evaluation of vendor information to assure no unexpected or unusual hazards will result in the purchase or use of the material.

19. Why aren't the following buildings included in Section 5.4.11.1.1.x that discusses the design bases for maximum fires and fire hazards?

- 2736-ZC loading dock (including transient storage)
- 2736-ZA diesel generator
- 234-5 south annex
- 234-5 west annex
- 2734-ZK acetylene and oxygen bottle storage
- 2734-ZJ liquid nitrogen storage
- 252-Z-1 transformer
- 267-Z sprinkler system building (including mechanical damage, e.g., truck impact)
- 2722-Z temporary tributyl phosphate storage (transient hazard condition)
- 2503-Z switch yard
- 2712-Z stack sampler diesel
- 2715-ZL drum storage
- 2902-2 water tank

Response. These buildings were not included due to their low dollar value and/or insignificant consequences. They do not provide functions to the main facility which are vital to maintaining the facility within the safety envelope described in the FSAR. As discussed in the response to General Safety Review Question 5, Westinghouse is performing a detailed review of FSAR implementation of the FSAR which includes validation of the systems, structure and components important to safety as described in the FSAR. Necessary configuration and operational controls for the above structures are included in the review.

20. Please provide a discussion that will resolve the discrepancies between Tables 5.1-1. and 5.4-13 of the uses/functions of the following buildings:

- 234-ZB
- 241-ZA
- 241-ZB
- 2701-Z
- 2722-Z (temporary tributyl phosphate storage)
- 2734-ZB
- 2734-ZF

Response. 234-ZB and 241-ZA. We see no essential difference in the descriptions in the two tables other than one may go into more detail than the other.

241-ZG. Facility 241-ZG contains the caustic tank and 241-ZB is the change room.

2701-Z. This building is an abandoned guard house that was used to store some instruments.

2722-Z. This truck unloading pad does not appear in Table 5.4-13 but will be added. It currently is being used to hold drums of tri-butyl phosphate.

2734-ZB. This is a semi-open building equipped with gas manifolds and bottle connection jumpers. It is no longer used for gas delivery but there currently are three bottles of argon chained in the racks.

2734-ZF. This is a semi-open building constructed to house gas manifolds and bottle racks. It is no longer used for gas delivery but is presently being used for miscellaneous storage.

The space provided in these tables for facility description tends to force the author to be terse. Thus some apparent discrepancies may not in fact be so.

21. **The status of fire alarm and fire systems for the following buildings were not found. Please provide a discussion that addresses this issue.**

- 232-Z, Incinerator
- 252-Z-1, Electrical Transformers
- 252-Z-2, Electrical Transformers
- 2701-ZA, Central Station Alarm Facility
- 2712-Z, Stack sampler power generators
- 2734-ZK, Gas bottle storage
- 2736-ZC, Loading dock

Response. Adding these buildings and facilities to Table 5.4.15 would add to the completeness of the FSAR but would not change any of the conclusions regarding risk of operating. Westinghouse will add this information in the next FSAR update. Please refer to response to Question 19 above.

22. **Use of the 50,000-gallon tank as a alternate water supply does not appear to meet the requirements of 6430.1A, which states that the light hazard conditions should not be used.**

- **The National Fire Protection Association lists a minimum of 30 minutes minimum duration for "light hazard" (the very lowest rating) and 60 to 90 minimum capacity for ordinary hazard.**
- **Tank water is also needed for system cooling, etc. (2-hour tank reserve), thus the combined water use gives a tank rating of 34 minutes.**

If the tank is not used for sprinklers and fire hoses, the references to the tank in the fire protection section should be removed to prevent the confusion. Also, the secondary water source should be more clearly identified.

Response. The high tank at PFP is one of three located in the 200-W Area that have a combined capacity of 200,000 gals. These tanks provide a minimal backup water supply. It is recognized that the three tanks do not provide adequate capacity. Project B-604 when completed will remedy this problem and the high tanks will no longer be needed for backup fire protection.

- 23. The safety classification of 234-5Z duct work (Table 4.4-2) was not clearly distinguishable.**

Response. Westinghouse agrees. Various components of the ventilation system were identified on the table but the duct work was not uniquely identified. The duct work is safety class 3.

- 24. What are the types and amounts of radiation sources in tunnels (especially on the mobility to ground water in event of flooding)?**

Response. The pipe tunnels contain liquid waste pipes. There is some residual plutonium contamination in the tunnels that has been painted over but the area does not require respiratory protection on a normal basis. Leaks are readily detectable and are repaired when they occur. Mobility of plutonium to the ground does not constitute an onsite or offsite dose problem.

- 25. Do the following buildings meet the National Fire Protection Association code?**

- **236-Z**
- **232-Z**
- **241-Z**
- **242-Z**
- **291-Z**
- **2736-Z**
- **2736-ZB**

Response. Yes. This comment refers to a partial listing of PFP buildings provided in Table 5.4.14. The table was included to provide additional information on auxiliary buildings not to serve as a complete list.

26. Should plutonium be included in the hazardous materials inventory (Table 5.4-17), including chemical toxicity?

Response. The Toxic Substance Control Act and the Hazard Communication Standards exclude radioactive materials managed under the Atomic Energy Act of 1954, and those Special Nuclear Materials managed under 10 CFR. All special nuclear materials stored or in use at PFP fall under the authority of DOE and is managed according to applicable DOE Orders (5480.10 and 11, and 5820.2A) not under the standards. A detailed inventory and hazard communication program already exists for SNM and the HazCom or SARA requirements would be redundant to the current system.

27. Please provide additional information on why potential filter failure is not considered due to smoke loading for the RMC fire scenario (9.2.2A.1.1, Page 9-315). A discussion should be included on the Lawrence Livermore National Laboratory Alvares et al. (1980) test and its application to the PFP filters and why it is used instead of the 1989 Westinghouse Hanford Company document (WHC-SD-EN-RA-001) that states "for the case of the RMC line, all filters would most likely rupture and could potentially exhaust passive material, toxic material, and small quantities of nuclear material to the environment."

Response. The copy of WHC-SD-EN-RA-001 read by the reviewing organization was a draft. In an unfortunate "passing-in-the-mail" type incident the reviewers did not see the approved copy. The approved copy makes extensive use of the Alvares et al (1980) data to show that the filters would NOT be destroyed.

28. What is the potential for toxic chemical releases due to fires, explosions, and other accidents at:

- 2735-Z
- 2736-ZC?

Response. The 2735-Z facility consists of two, 5,000 gal tanks containing nitric acid and aluminum nitrate nona-hydrate (ANN). These chemical do not explode or burn. The tanks are held in a concrete basin. The wall of the basin stands approximately 16 inches above grade. The adjacent pad for unloading trucks has a series of heavy steel pipes set on the east side and a guard rail. Thus a direct impingement of a truck on the tanks at yard speeds would be quite unlikely. Vehicle fires in wrecks even at highway speeds are rare and much more rare at low speeds and explosions very rare. A nearby vehicle fire could heat the contents of the tanks but would have to be

burning for a considerable time in relation to fire suppression in order to start boiling these volumes of liquid even considering that they would not always be full.

A seismically induced release of nitric acid is analyzed in Section 9.2.4A.6.5.3.E. As a nitric acid salt the ANN would have a less effect.

The 2736-ZC Building is metal framed and sided this is used for storage of empty Department of Transportation (DOT) radioactive materials shipping containers. There is no potential for a release of radioactive materials or chemicals from this building.

29. **The metal lathe and plaster walls for 234-5 and 236-2 were identified as non-flammable, but are they considered fire resistant, and, if so, how was the fire rating obtained? Without knowing the composition of the plaster walls, the effects of fire, water spray, water submersion, and long-term moisture degradation on plaster walls could not be determined. The safety (and significant non-safety) functions, the lathe, and plaster walls performance (e.g., filter room wall boundary), and the wall's response to accidents need to be identified.**

Response. It is not clear where the statement "non-flammable" is used. In Section 9.2.2B.2 there are discussions of the plaster on metal lath walls and ceilings in the second floor, non-radiation zone. The discussions state that without detailed knowledge of the materials and mode of construction it is impossible to assign a true fire severity rating. The discussion points out that even though they are not rated they would have a real effect in slowing the progress of a roof fire specially with the generally low combustion loadings and a Class 1 roof.

There is no reason to think that the walls have deteriorated as there are no water sprays and the moisture levels are not such that gross deterioration of the walls would take place. These are walls in offices and corridors and store rooms. The filter room walls for Zone 4 may have deteriorated but the argument presented did not quantify the degree of mitigation provided. The fire loading next to the filter room walls is extremely low.

In an early September meeting with PNL we were asked what the effect of fire sprinkler water would be on the plaster walls of the filter rooms. Our Plutonium Process Support Laboratory did a study in which they sprayed water on a section of plaster removed during a remodeling effort in a locker room. The plaster here is considered to be the same as was used through out the facility as prints show no differentiation. The spray test showed no significant deterioration of the plaster after several hours of spraying. The study was documented in a letter report and a copy was sent to the PNL reviewers.

In the earthquake we assume that much of the duct-work fails and this does not exempt the filter rooms themselves. A plutonium release from the filter rooms is assumed.

30. **Is the SAR intended to meet the fire protection analysis report requirement of DOE 6430.1A, 110-6.2? If so, then it should be clearly stated, and, if not, the document meeting this requirement must be referenced.**

Response The DOE Order revision of this order, which is currently in draft form, has renamed the analysis a "Fire Hazards Analysis". This analysis has been prepared for PFP and is referenced in the FSAR.

Neither of the scenarios in Section 9.2.2 are intended to meet the requirements of DOE 6430.1A. The fact that the fires are finally put out by the Fire Department seemed to make the distinction clear. It is no accident that the scenario is very similar to the design-basis-fire format because it was an attempt to show the probabilities that might exist and to take it far enough to have a plutonium release. If the scenario had been truncated or allowed to burn to completion the probabilities of all elements would not have been illustrated.

The original 6430.1A analysis was done as Appendix B to SD-CP-ES-039, Rev 2, T. Romano, "PF-Plant RMC Line Panel Replacement Engineering Study", February 1985, Rockwell Hanford Operations, Richland Washington. A modification was made to this study in the form of an attachment to Internal Letter 12900-86-077, R. A. Huckfeldt to J. J. Roemer, "Design Basis Fire from Remote Mechanical C Line", January 31, 1986, Rockwell International. The version in the letter addresses the use of Rooms 230-A and 230-B as part of the RMC-line operation. It is not clear that these should be documented in the FSAR.

The references in Section 5.4.11.1.1 to Chapter 9 and to WHC-SD-EN-RA-001 will be reviewed and modified as is probably necessary.

31. **Please provide a discussion on the control of fire protection water runoff that is contaminated with hazardous chemicals and/or radiological (compliance to DOE6430.1A, 1304-7.2).**

Response. A brief statement in Section 5.4.11.3.2 states that there are no provisions for runoff of fire suppression water. Such water would not pose a threat to the safety of people on-site or off-site.

32. **Does the exterior design of any of the facilities follow the recommendations of National Fire Protection Association 80A (DOE 6430.1A, 110.6.4)?**

Response. Yes, this is covered in Table 5.4.14.

33. **Has the potential for an explosion, in a glove box or other primary confinement space, to cause the light fixture and associated barrier to be violated been considered? For example, has the potential for a glove box explosion to rupture the light fixture area barrier and cause release to the above space, including potential impact on the heating ventilation and air conditioning ducting above, been considered? Consider Pages 9-268 and 9-269.**

Response. Any explosion in a glovebox will be likely to have an effect on the glovebox light fixture, generally mounted on top of the glovebox. Larger explosions such as that described on Pages 9-268 and 9-269 will have an effect on the ceiling above the glovebox including the room light fixtures. The ceilings in the RMC-line rooms are approximately 6 ft to 10 ft above the top of the gloveboxes but the explosion cited extends the damage into the duct level above as described at the bottom of Page 9-269. Damage to ducts is implied by exception as the text states that filter boxes are not damaged because they are on the floor above.

34. **Page 9-320 states that an "avenue of investigation" will be performed later. However, it is not clear what the avenue of investigation consists of and what the significance is to this facility safety analysis report section.**

Response. Please see Section 1.5 "Requirements for Additional Technical Information", Section 1.5.2 "Physical Strength of HEPA Filters" It is speculated old filters become very structurally weak then only a slight smoke loading will cause them to rupture. Our type of HEPA filters are shown in the manufacturer's literature as having an ability to withstand a differential of 19 in. water while our fans pull only about 6 inches water (Figure 9.1.7A-4). Thus there is a considerable tolerance for degradation which is probably why there is little or no manifestation of the potential problem in the industry.

35. **A list of all possible sources of ignition in the PFP facility should be provided (Section 9.2.2A.2.1). The probability of not noticing the fire should have a quantitative basis.**

Response. A list of all specific sources is not practical to provide. In general, any electrical wire, rotating or reciprocating machines, motors, incandescent bulbs, welding equipment, furnaces, or soldering guns could be a source of ignition given that an appropriate ignitable material was present. The most obvious heat sources in the area with highest fire loading are the electric furnaces that surround the calciner and fluorinator tubes on the RMC-line.

There is no way of getting data on the probability that someone will not notice a fire. It can only be an engineering estimate based on the authors many years experience in industry as well as the combined experience of those plant personnel who were

consulted and those who reviewed the document and agreed. Swain and Guttman (1983) give some guidance but nothing definitive.

36. **The C-6 response time for the Hanford Fire Department should be broken down for more detailed analysis of the sequence of steps involved in their response to establish the value.**

Response. A branch of a tree consisting of nothing but engineering estimates would be no better than a combined estimate given that a reasonable argument was presented to support the conclusion.

A.6 SEISMIC

1. **Please describe which glove box modifications are required prior to startup (Section 9.2.4A.2).**

Response. The glovebox that require modification prior to restart is HC-4 in room 166. This has been completed. There are also upgrades required on filter boxes 14, 21 (both completed) and 9B (work started 7/92). Additionally, filter boxes 7A and 9A need to be cleaned out (this removes the Pu and, thus, prevents any release. These filter boxes exhaust the RMA line which is no longer used). The E-4 duct between filter boxes 9A and 9B will also be upgraded prior to restart pending its removal as part of the Pu remediation effort. The significance of these upgrades was reevaluated subsequent to the publication of the FSAR and is documented in WHC-SD-C084-RA-001 which was provided to Pacific Northwest Laboratory for information. Additionally, cable braces on the water tower are placed to prevent the tank from crashing onto Building 270-Z which is directly east of the tower. The cables extend nominally north-west and south-west from the tower. The height of the tower is less than the distance to Building 234-5Z so bracing was not needed in that direction. In addition, observation of water tower damage indicates that water tanks fall straight down within the span of the legs (**Earthquake Engineering**, 1970, Prentice-Hall, Page 199)

2. **Please show how additional assumptions are to be maintained valid and clarify the meaning of the last sentence (Section 9.2.4A.2.2).**

Response. There is no need to show that the additional assumptions are maintained valid. The first two are statements that the analyst considered the worst case conditions that could be induced by random failures of components. If the assumptions are not "maintained valid" lesser consequences result. These are enveloped by the worst case scenario conditions assumed. The third assumption centers around the combined probability of an abnormal situation occurring in coincidence with the seismic event. It is used to form the basis to start the analytical process of the seismic event from normal processing conditions rather than assuming a

pre-existing abnormal situation. The last sentence summarizes the outcome of utilizing the three assumptions in the development of the analysis.

3. **Where are the events described in Sections 9.2.4A.2.2D and 9.2.4A.2.5 evaluated?**

Response. The events are described as follows:

9.2.4A.2.2D Pages 9-469 and 9-478.

9.2.4A.2.5 Pages 9-405, 9-479, 9-509, and 9-536.

4. **Please indicate whether modification to separate the floor and columns has been completed (Section 9.2.4A.2.13).**

Response. These modifications were not done and are not required as explained in WHC-SD-CP-SA-017.

Response to 10/2 Review: Failure of the concrete, Column Line 8 wall in the Duct Level is a Safety Class 3 event as no Pu is released. The draft document on the subject of what to do with the wall is being reviewed at this time. The author recommends that no work be done on the wall. With re-alignment of rooms and passageways occupancy of the spaces where the wall might collapse will be quite rare. Collapse of the wall would not open any new release paths. In addition, energy absorbed in breaking the wall would tend to protect the rest of the building.

5. **Section 9.2.4A.6.1 indicates that glove boxes are assumed to remain upright. This is inconsistent with earlier Section 9.2.4A.2.12, which indicates the glove box stability has been calculated.**

Response. There is no inconsistency. The analysis was done but most of the gloveboxes required upgrades. The FSAR is assuming that these upgrades are not done. This is the correct assumption. The document detailing the response of most gloveboxes in the PFP is, E. C. Ocoma, WHC-SD-CP-SA-007, April 10, 1990, "PFP Safety Systems and Components Seismic Evaluation-Phase II." A detailed study was made to determine which gloveboxes needed to be upgraded and which did not. The decision was based on changes in the operating mode of the plant and upon cost-benefit (dose reduction) and documented by K. R. Herzog, WHC-SD-C084-RA-001.

6. **Pages 9-380 through 9-395 should describe the aggregate affect of the seismic event on the 234-5Z building to assess if large or small gaps are opened in the panels that would provide an inside-to-outside release path for any potential spread of radioactive materials from failures inside the building.**

Response. The description addressing building panels on Page 9-380 indicates that the panels are very rigid in diaphragm action and do not drift when the building shifts into

essentially a parallelogram shape. The drift is only enough to cause some elongation of bolt holes. The folded connection between panels, designed for weather tightness, would prevent gaps as the folded parts could extend a considerable amount without separation.

7. **Please provide additional information to support the basis for stating that resuspension of particles from an enclosed duct failure in an earthquake should be less than that from a tipped over beaker (Section 9.2.4.0.2). The rate of particle resuspension from a tipped over beaker as conservative should be substantiated.**

Response. The brief description of the powder spill experiment given in Section 9.2.4.0.2 might give a reader the impression that the experiments consisted of simply tipping over a beaker on a table. The detailed calculations presented in Section 9.2.4A.6.3.10 use the energy relationships presented in Halverson (1986). This work used, in part, data from another PNL document by Sutter, Johnston and Mishima entitled "Aerosols Generated by Free Fall Spills of Powders and Solutions in Static Air", NUREG/CR-2139, PNL-3786. The title of the document is much more descriptive. Beakers containing various compounds were hung in the center of a large test chamber and then completely inverted remotely while samplers mounted at many locations measured the resulting suspended particles. One of the compounds used was very fine, ball-mill ground uranium oxide powder.

The ducts in question are a mixture of 6" and 4" diameter, schedule 10, stainless steel pipe that is not expected to be squashed with a large swoosh of air as might be postulated with typical galvanized or aluminum, sheet-metal ducts found in ordinary air handling systems. We assume that the ducts (pipes) would be torn apart or open and that the contents, assumed to be dry powder, would spill out. The remaining ducts are down-stream from a caustic scrubber system and the plutonium is most likely strongly fixed to the steel. It stretches credulity to assume that the old residue, in a duct located down stream from a caustic scrubber would contain loose plutonium oxide powder with a particle distribution of respirable size particles anywhere near as high as was used in the experimental powder. Yet without any physical indication of the nature of the residue Westinghouse was obliged to use the conservative assumption.

8. **In order to perform a non-linear dynamic structural analysis for the PFP, Blume (1987) developed time history curves for use with the finite element model. These time history curves were generated by starting with initial time history curves and modifying them with an iterative procedure until they closely matched the Hanford Site response spectrum curves. Please demonstrate that the derived time history curves result in the worst case building response.**

Response. Several documents have been reviewed in order to determine the requirements for performing linear and non-linear time history evaluations. The documents reviewed were:

- NUREG 0800 (1981) Section 3.7.1 and 3.7.2; Requirements document for NRC governed plants (non-binding for DOE plants) that was in effect in 1987 at the time of the Blume evaluation. No mention of multiple or worst case time history curves is made. Time history curves for the three orthogonal directions are to be independent and should generally envelope the design response spectrum. This was done. The 1987 time history curves were checked for enveloping the design response spectra.
- NUREG 0800 (1989) Section 3.7.1 and 3.7.2; Requirements document for NRC governed plants (non-binding for DOE plants) that is in effect now. Use of a single time history curve is justified by satisfying a target power spectral density requirement in addition to the design response spectra enveloping requirement. Otherwise, multiple time history curves may be used. The results from the several time history evaluations shall be averaged for the final resulting loads. This revision was not in effect at the time of the 1987 evaluation. A single time history curve that had not been checked for power spectral density was used in the 1987 evaluation. The 1987 time history curves were checked for enveloping the design response spectra. The 1989 evaluation developed a new set of statistically independent time history curves which satisfied power spectral density requirements.
- DOE 6430.1A; Requirements document for DOE plants. References UCRL 15910 for seismic requirements.
- UCRL 15910; Seismic requirements document for DOE plants. References TM-5-809-10-1 for dynamic evaluation techniques.
- TM-5-809-10-1; Dynamic seismic evaluation technique document. Provides guidelines for response spectrum evaluations. Recognizes the need to perform time history evaluations for buildings that respond in a non-linear manner but does not provide guidelines for time history evaluations. Instead, an approximate response spectrum method is presented.
- ASCE 4-86; American Society of Civil Engineers Standard for Seismic Analysis of Safety-Related Nuclear Structures. Not a required document but an industry recognized standard for seismic evaluations. For linear-elastic evaluations, no mention of multiple or worst case time history curves is made. Time history curves for the three orthogonal directions are to be independent and should generally envelope the design response spectrum. This was done. The 1987 time history curves were checked for enveloping the design response

spectra. For non-linear evaluations, more than one set of time histories should be used and the results averaged. The response of the structure did go into the non-linear range at four out of the 627 structural steel connections, however the structure as a whole responded essentially as a linear elastic structure. Matching a single time history curve to the design response spectra is considered sufficient to meet the seismic input criteria.

- Two independent reviews of the seismic evaluation were performed.
- EDT R-0949; Review of the Seismic Evaluation for the 234-5Z Building at the Hanford Site, J. D. Stevenson & Associates, December 1987. According to Dr. Stevenson, *"I have restricted my review and comments to general analytical assumptions, methodology, and conclusions of the reports."* Dr. Stevenson further stated *"This report is without a doubt one of the best structural-seismic analysis to failure reports I have ever seen."* Nowhere in this review document does Dr. Stevenson mention the use of multiple or worst case time histories or the lack of use as being a shortcoming.
- Memo APL88-015: Robert C. Murray (Lawrence Livermore National Laboratory) and Stephen A. Short (Impell Corporation) to James R. Hill (Office of Nuclear Safety Headquarters, U.S. Department of Energy) dated January 14, 1988. According to Dr. Murray and Dr. Short, *"It is shown in Figures 16 through 18 that the acceleration time history used closely matches the SSE response spectrum. However, the acceleration time histories shown in Figures 19 and 20 have a single large peak at 0.25g but repeatable peaks at a much lower level (on the order of 0.15g for east-west motion and 0.10g for north-south motion). Since the response of this structure to SSE motion was essentially linear elastic, matching the response spectra is sufficient to meet the input criteria. If there had been a large amount of nonlinear beam column rotations or other inelastic behavior, it is not clear that these acceleration time histories would conservatively meet the criteria since repeatable peaks of higher amplitude may have been very detrimental producing increasing inelastic deformations (especially, in the north-south direction which did not have the resistance of concrete walls). If there had been more significant nonlinear behavior, we were told that URS/Blume would have repeated the analysis for several independent earthquake time histories."*

The derived time history curves do not result in a worst case building response, but, there is no requirement to do so. By current standards, the use of a single set of time histories that matches the design response spectrum is appropriate for use in linear elastic evaluations. The URS/Blume evaluations are considered appropriate for the results obtained.

A.7 CHEMICAL HAZARDS AND EXPLOSIONS

1. The Section 9.2.4A.6.3.1.0.a assertion that the explosion over pressures of 700 psi and 30 psi (bottom of Page 9-435) will not fail the concrete walls is unpersuasive. On following the cross-reference to Section 9.2.1E.6, we see the calculation of two parameters and the assertion that these parameter values are shown in the reference NDRC (1946) to imply no failure of the 8-in. concrete wall. The reactor containments fail at 50 to 150 psi range and are tied together at floor and ceiling and can be expected to be stronger than the Room 228 walls which are tied together at the floor but not the ceiling. Further, the resistance of a reinforced concrete wall to bending, shearing, and overturning is going to depend not only on the thickness, but on the details of design of the reinforcing steel.

Finally, W. E. Baker, in a paper "Blast Pressure Effects: An Overview" in the collection Design Considerations for Toxic Chemical and Explosives Facilities, Scott and Doemeny, Eds., American Chemical Society, 1987, has a graph (Figure 24, Page 28) for Damage to Reinforced Concrete Walls caused by Detonation of Uncased Explosives as a function of scaled distance and scaled wall thickness, which looks like it might be equivalent to the information in the NDRC (1946) reference. Duplicating the calculation of SAR Pages 9-268 and 9-269 and using the Baker graph leads to the conclusion "minor or no damage." However, the types of damage referred to in the Baker paper are cracks and craters on the explosion-facing side of the wall, spalling from the other side, and perforation of the wall rather than the issue of gross structural failure of the wall due to blast over pressures and impulse loadings. Therefore, this type of calculation does not seem appropriate for gross structural failure of the Room 228 walls. Please provide a more appropriate calculation of gross structural damage for this analysis.

Response. The comment is a good comment in that it points out the problem with using references. The basis for the conclusion that the walls in room 228-B survive the hydrogen explosion originating from failure of glovebox HC-13MD was as you stated (the NDRC and Baker reference). It is also acknowledged that other factors influence the survival of the walls. Some of these are mentioned in the comment. It should also be pointed out that there are factors that appear to show that the conclusion may not be incorrect. These are the fact that the ceiling and the wall to the side are both determined to fail. If these walls fail fast enough, they could act as a rupture disk, possibly relieving pressure fast enough to result in the survival of the concrete wall.

Regardless of the above, on page 9-438, it is stated that the consequences are severe enough that modifications will be made so that the explosion will not occur.

With regards to the hydrogen explosion in the fluorinator (Section 9.2.1E), again, it might be argued that the failure of the ceiling and the wall to the west will result in pressure relief such that the concrete walls will survive. If it is assumed that the concrete walls fail in the vicinity of the explosion, the consequences will change only a little. The change will be that the value for "V" on page 9-270 will be larger so that the value for "grams per cubic meter" at the top of page 9-271 will be less. This will lower the value for "a" which will make the mass released term (now 0.07M, on page 9-272) larger. The release will still be filtered so that release, while larger, will still be well within the Risk Acceptance Guidelines. Since the hydrogen explosion potential in the fluorination is eliminated with the discontinuance of plutonium metal production discussion of this scenario is now moot.

As discussed under Fire and Explosions, Question 8, Westinghouse agrees that the scenario for the acetylene explosion is in error, the 8 in. thick concrete walls would not contain the pressures from an explosion of a room full of an acetylene-air mixture, and acetylene use will be essentially eliminated.

2. **It should be noted that the release of acetylene to the room and the subsequent deflagration will result in the buildup of pressure in the room until either the peak pressure is reached, normally about eight times the initial absolute pressure, or else the room confinement fails. It does not appear that this situation is represented by the equations presented in Section 9.2.1F. The explosion accident analyses in Section 9.2.1F uses an equation from Gungan (1979) to determine the TNT equivalent for an acetylene vapor cloud explosion. The same equation is used in Section 9.2.4A.6.3.1.F to determine the TNT equivalent for a propane explosion. In this review the equation has been rederived, correcting a units error in Gungan (1979). In both sections a very low efficiency and, thus, a very low TNT-equivalent mass is calculated, leading the analyst to discard the accident sequence as inconsequential. There are a number of actual or potential problems, however, with the use of the Gungan equation.**
 - **It is not clear to the reviewer that the variables "epsilon" and R have the same meaning in the SAR application of the Gungan equation that they do in the original.**
 - **The values of flame velocity, "nu," used in the SAR are initial deflagration flame velocities. The Gungan equation uses "nu-bar," the average flame velocity from ignition until conversion of the blast wave to a shock wave, after flame acceleration and just prior to conversion of deflagration to detonation.**
 - **Gungan writes (Page 126): "The application of either of the above expressions for efficiency is not recommended; they merely serve to**

illustrate the much more complicated form of UVCE yield, expressed as TNT, than has heretofore been presented."

- Guban provides data on 100 actual UVCE (unconfined vapor cloud explosions), including, where possible, the amounts of flammable material involved and the TNT equivalent (where TNT equivalent is defined as the amount of TNT that would produce a shock wave resulting in the same damage as the actual UVCE). Figures 1 and 2 of Guban are scatter diagrams of the percent yield efficiency plotted against mass of vapor in the cloud (Figure 1) and an arbitrary combustion figure of merit (Figure 2). There are no obvious correlations in the scatter diagrams. The percent yield efficiencies range from .02 to 32 to 65% (32 to 65% means that the damage from an explosion of 64 ton of propane produced an UVCE with damage equivalent to that of the detonation of 20 to 40 ton of TNT).
- Guban writes (Page 104): "In no case, based upon current data, can a blast efficiency (related to TNT) be predicted with any confidence whatsoever from a knowledge of the flammable material and the likely quantity escaping into the open air."
- Guban writes (Page 106): "TNT equivalence has led to the idea that UVCEs produce blast waves having essentially the same form and proportions as explosive shock waves. There are absolutely no grounds for such similarity, either in conception or in reality. Theory predicts that a large pressure wave will tend to steepen on the leading edge and become more shallow on the trailing edge...Whereas, the pressure levels from a UVCE and its 'equivalent' TNT counterpart may be similar, the energy content of the UVCE pressure wave may be many times greater (reflecting the intrinsic energy difference between TNT detonation and vapor combustion). The only way this additional energy can appear, unless it disappears or somehow is not released, is in extended pressure waves--necessarily of long duration."
- Guban writes (Page 122): "Here, then, we see the principal difference between UVCE blast waves and explosive shock waves. The energy in blast waves is relatively low grade--in pressure terms--but, because of the long durations of the pressure pulses, damage often results which is not dissimilar to that produced by conventional explosive shock waves of much shorter durations and higher pressures."

For these reasons, the contractor should review and revise the calculations used in Sections 9.2.1F and 9.2.4A.6.3.1.F to determine long-duration pressure pulses and their consequences in these accident scenarios.

Response. Westinghouse agrees that the scenario for the acetylene explosion appears to be in error and that the 8 in. thick concrete walls would not contain the pressures from an explosion of a room full of an acetylene-air mixture. Please see the response to comment 8 of the Fire and Explosion section.

3. **Many of the tanks in the PFP have concentration limits (reference Section 8.2.1.1). How are the concentrations monitored and how are the tanks prevented from exceeding the concentration limits?**

Response. Concentrations are monitored by a combination of direct sampling and calculation based on upstream conditions and flows in addition to monitoring tank weight factors and specific gravity. Tanks in the PRF are rarely operated near the 450 g/L limit for process control purposes. Criticality limits are implemented through adherence to operating procedures. Strict limits are imposed on the vessels outside of PRF into which solutions must be transferred. Thus transfer source tanks must be sampled before transfers are made.

4. **Will Glove Box HC-15 in Room 228B of 234-5Z be operated? If so, will molten plutonium be present in current operations and how does this impact plutonium releases if exposed to air or water (moisture)?**

Response. No. In the new stabilization mission the old RMC-line process will be shortened. We will make PuO_2 in the calciner as before and then further oxidize that powder in the present fluorinator used as a second stage calciner and send it via two conveyers to an end-stage glovebox for packaging. The majority of the metal production gloveboxes will be by-passed.

5. **The procedures require an operator to perform a surveillance every 2 hours on the aqueous solution leaving the CA column. This surveillance is used to detect abnormal operation in the CA column and depends on the operator noting a change in color. Is this method still effective if the observer is color blind? If not, does this method still have value? Do procedures exist to prevent color blind individuals from working on this process? Is the CA aqueous solution collected and monitored before being sent to the 241-Z tanks?**

Response. One of several indicators of an upset in CA Column chemistry is to examine the color of the solution that can be seen in the glass, disengaging section. Probably even in shades of gray one could see changes in the intensity of color due to increased plutonium concentration. There are no procedures in place to prevent a person with inability to see blue from being assigned to this task. The primary control on plutonium concentration in the holding tanks consists of sampling and analysis.

6. **How many and what are the specific forms of feed stock chemical to the plutonium reclamation facility? To the RMC line? Please describe the specific**

forms of material and their matrix, e.g., PuO_2 in broken slag and crucible parts via the MT-5 dissolver.

Response. An understanding of the feed stock for the two facilities can be obtained by reading Sections 6.1, 6.2, and 6.3.

In general the feed for the PRF will consist of materials that contain water or organic matter that would generate gas if stored in sealed cans in the vaults. Solids will be dissolved in Glovebox MT-5 and sent to PRF as impure Pu nitrate solutions in nitric acid. Liquids that are stored as impure solution in shipping containers will be loaded into PRF where they will be both purified and concentrated.

Feed for the RMC-line consists of Pu nitrate solution in nitric acid with little impurity.

Slag and crucible (S&C) parts were to have been dissolved in the S&C dissolvers but may well be stabilized in a new process that is under study and development. There would be far too much gas generation to dissolve this material in MT-5.

7. **What hardware or procedural systems have been put in place to deal with the possible hydrogen buildup in plutonium nitrate storage containers which have been stored (static) since the processes were last operated (several years)?**

Response. The containers are vented. PR cans have vent clips installed under the lids within 13 days of filling and the L-10 type cans have the vent valve opened. These actions are controlled by established operating procedures.

8. **Has an analysis been performed regarding radiolytic hydrogen formation in the RMC feed tanks and permeation reduction factor product tanks?**

Response. Yes, please see Section 9.1.8.4, Page 9-181 and Section 9.2.4A.6.3.2.A.n, Page 9-460.

9. **What safety system(s) is in place to prevent an explosion in the hood in Room 144? This would be caused by an abnormal event resulting in lack of ventilation and propane as the fuel.**

Response. The primary safety provision is that only nominally 1 lb bottles of propane be used. Plutonium is seldom measured anymore by depositing material on a disc or planchet. For those few times when it is, a 1 lb propane bottle, attached to the outside of a hood in Room 144 is used. The gas is brought into the hood by a tube to a small burner that is lighted and then extinguished after each use. Should the flame go out the normal hood ventilation would sweep the gas away. The face velocity of the hoods is maintained at greater than 125 ft/min so with two arm-ports the volume is

about 250 ft³/min. The bottles contain about 8 ft³ of propane (Section 9.2.4A.6.3.1.F, Page 9-421). The control valve is on the bottle on the outside of the hood, so full flow is not available to the glovebox. An accident involving a broken spigot on the bottle during an earthquake is described in the above reference.

An explosion during a state of no ventilation was not analyzed. In order to have an explosion one has to combine the probability that the electric and steam driven fans fail, and that no natural draft was operating at the stack. Also, that the seldom used burner was turned on but not lighted, that the Technician failed to shut it off before evacuating, and that there was an ignition source. Then for the resulting explosion to be significant there would have to be a rather sizable inventory of plutonium in the duct and filter box, which there is not now, and that the explosion be great enough to breach the building. The rooms through which the ducts run are very large which dissipates explosive energy.

10. **Have the consequences of using oxygen rather than hydrogen-fluoride in the fluorinator (Sections 9.2.1E and 9.2.4A.6.3.1.0) been examined in relation to the accident scenarios, e.g., fire? Please discuss this issue further.**

Response. It has been decided that pure oxygen will not be piped into the fluorinator of the RMC line. Air will be used instead to oxidize the Plutonium.

11. **Have the consequences of an acetone fire been considered (Sections 9.2.10.2.3 and .4)? What are the consequences of an explosion involving flammable fumes in the concentrator or exhaust duct work?**

Response. No. Acetone is a seldom used chemical. At one time it was used frequently for cleaning and drying laboratory ware. Environmental regulations have severely limited its use. It was purchased in one quart metal cans and volumes were strictly controlled from a fire safety provision.

- 12.a **How was the probability of 3.8×10^{-15} arrived at in Section 9.2.1A.1? A cursory analysis of the fault-tree as it stands seems to indicate a human error probability of 4×10^{-4} . This question could be asked of almost all the human error estimates. The dependencies among sequential errors is just not being considered. This analyses raises several related questions. Where did the probabilities for the human error rates come from? How were the numbers modified based on plant specifics? How were specific human errors identified for inclusion in the fault trees at the exclusion of others?**

Response. Operator failure rates were assumed to be 1E-2. See response to comment 1 under the heading of Probability. The sequence in question contains three operator errors. Probabilities for these kind of sequences usually contain a factor to include common cause, but in this case it is not justified. Event EF3 occurs prior to the flood

and is independent of it. In this case the column is working fine but the operator has mistakenly allowed for hold up of solution in Tk-43, by valving out the discharge. Event EF7 is related to EF3, in that once the operator notices the error (level increasing in a flow-through tank), he notifies the supervisor. The supervisor then mistakenly orders continued operation. An explosion cannot occur under these conditions. A column flood must also occur at the same time that the operator has valved out the discharge. The flood would be noticed regardless of whether the discharge is valved out. Paragraph 4 on page 9-199 shows the numerous indications of a flood. It is, therefore, believed that the operator errors are different enough in nature that they can be considered independent.

In a series of calls with the reviewer we found that he was concerned in a generic way with any scenario that contains several items of human failure and has resulting probabilities much below $1 \text{ E-}05$. His reasoning is that there is usually some common mode failure such as evacuation of a control room that negates all the well intentioned human reactions. The nitrated organic (red-oil) explosions in the PFP involve human failures in different parts of the process at different times. It takes a series of oversights to accumulate the TBP over time and then a number of oversights, also spread over time, to let equipment deteriorate or to fail to recognize or respond to signals. Thus it is our contention that the probability of the scenario as presented is quite low. We do believe, however, that failure of the basic assumption presented in Section 9.0.1.2.4.4 concerning good quality in conduct of maintenance is probably over-riding. But these considerations are true in all facilities in the world.

During a 11/10/92 meeting PNL personnel stated they wanted to confirm the explosion analyses on the filtrate evaporator presented in SD-SQA-TI-013, REV 0 that was performed in 1987 in support of a PFP restart. During review of the material in SD-SQA-TI-013 by Westinghouse personnel it became obvious that the document did not contain enough information to confirm the analysis. A number of drawings were copied and sent to PNL to be used by the review team.

12.b How were the numbers modified based on plant specifics?

Response. With regards to human or operator errors throughout the analysis of the PFP facility, a operator failure probability of $1.0\text{E-}02$ was used for errors of omission, $1.0\text{E-}03$ for errors of commission, and $1.0\text{E-}03$ for other errors. Errors of omission imply that the operator is aware of a procedure or out of tolerance condition, but does nothing about it. Errors of commission imply that an operator takes unrequired action and as a result causes an error. In such calculations, there is no time period considered. If the general philosophy defined in Human Reliability Analysis by Dougherty and Fragola was followed, there would be a time interval in which the action needs to be taken. Generally in process plants there is a significant amount of time available. According to Dougherty probabilities smaller than $1.0\text{E-}04$ would be used in these cases. The $1.0\text{E-}02$ used in the PFP analysis would be a very

conservative screening value and when the overall accident sequence still remains to be less than 1.0E-06 per year, no further analysis is required beyond screening.

12.c How were specific human errors identified for inclusion in the fault-trees at the exclusion of others?

Response. No plant specific data records included in the human error probabilities for the PFP SAR analyses. Human error failure rates are not kept at various Hanford site facilities. The best that can be done in this area is to query the operations personnel. This was done in the case of the CC column carry over of organics (4×10^{-4} /batch). The reasoning behind this rate is discussed on page 9-207.

The reviewer should be more specific with respect to the human errors that have been left out of the analysis.

13. Shouldn't the hydrogen generation from dilution* be considered as well (Section 9.1.8.4)?

* A telephone conversation with the author of the comment revealed that the word dilution should have been dissolution.

Response. No metal dissolution is planned so loss of ventilation during a gas generation phase will not happen. The Slag and Crucible dissolvers are not expected to be operated. Even if they were to operate the offgas would discharge into the large space that makes up the process cell and would be diluted (Section 9.2.1G.8).

14. Shouldn't the chemical exposure risk also consider the risk to workers in addition to the general public (Section 9.0.2.3)?

Response. No, because the purpose of the document (See Regulatory Guide 3.39) is to evaluate the results of accidents to offsite and onsite personnel.

15. Have hydrogen explosions in the duct work been considered? Please discuss the possibility and consequences of this accident scenario.

Response. Hydrogen explosions have not been considered in the duct work. The hydrogen explosions were considered to occur in areas where hydrogen potentially would be generated and accumulate, where there is an obvious ignition source and in areas where the explosion will take place in close proximity to the radioactive material. While it may be theoretically possible for hydrogen explosions to occur in the duct, the consequences will be less because the quantity of dispersible material is less. The scenarios in which hydrogen is assumed to be released into gloveboxes took place on the RMC-line as a result of metal production. With deletion of this process the sources of hydrogen are also eliminated.

A.8 HEATING VENTILATION AND AIR CONDITIONING (HVAC)

1. **In Section 11.3.5, an operational safety requirement has been developed for the continuous air monitor and associated alarms for the 291-Z-1, 296-Z-3, 296-Z-14, 296-Z-6 stacks during all building operating conditions when exhaust flow is occurring; however, continuous air monitors that monitor room exposure are not covered in the operational safety requirements. Since room continuous air monitors are credited with the detection of spills and loss of confinement events, shouldn't they also be bound by the operational safety requirements?**

Response. No. Please refer to Table 9.1-1 in the synopsis for events 9.1.1, 9.1.2, and 9.1.3. Note that these very conservative events would just barely exceed the criteria for establishment of an OSR even if there were no HEPA filters. The OSRs are used to detect or mitigate releases that would be significant on-site and off-site, not for events within the plant.

2. **The RMC line glove box air is continuously sampled and analyzed for moisture content by a condensation dew point hygrometer, which is set to alarm when the dew point rises above an established limit. What is this limit and how much margin does this give before over pressurization results?**

Response. The alarm is set at -10°F. Dew point is a measure of humidity and not pressure. Before PuO₂ is canned a sample is taken. Before the can is placed in storage the sample must be analyzed for moisture and be below 1% to prevent possible container pressurization (Section 9.1.4). Keeping the air dry was also a consideration during reduction of PuF₄ to metal as discussed in Section 9.1.6 but that was before our change in mission.

3. **In the analysis for ventilation upset conditions, it is assumed that the ventilation dampers fail in an open/closed position. Are the dampers tested to ensure that in fact they do fail in the correct position? What happens if they fail in the wrong position? Have HVAC damping sensor tip failures been modeled as a contribution to the release of radioactive materials in analyzed accidents? Is there a possible common mode failure for these sensor tips?**

Response. Failure in full open or full closed position gives the worst case. We do not assume that dampers fail in the safe position to initiate an event. The building complex with the most potential for radiation effects from a ventilation upset is that composed of Buildings 234-5Z, 236-Z, and 242-Z. During tests of the seismic shut-off switch, proper function of dampers is verified. In addition there are normally several times per year when conditions such as high wind make it necessary to take the system down to steam turbines. During these times the damper positions are not inspected for position but the air balance in the building goes to the correct configuration so one can surmise that the dampers are performing adequately.

We have not examined sensor tip or other control equipment failure per se but the failures that we hypothesize due to valve position errors will do the same kinds of things and leave the operating personnel in at least as large a quandary. The pressure effects would be the same and the recovery would ultimately be to drop all electric fans and use only the steam turbines.

4. **It is stated in Section 9.1.7A.1.2 that there does not appear to be any documentation for the numerous small valves associated with the ventilation system nor are there any labels explaining the functioning of the valves. During the meeting on June 23, 1992, it was stated that a program to label these valves had begun. Please provide information on the program to label valves in the ventilation system and the current progress of this effort. The lack of clear labeling could also result in an increased probability of failure due to operator actions. How has this been evaluated?**

Response. The labeling has not been completed. The fact that labeling is not complete will not increase the probability over what is described in the FSAR as no labeling had been done at that time. As discussed in the response to General Safety Review Question 5, Westinghouse is doing an extensive review of implementation of the FSAR as part of the current facility restart program. This effort includes validation of systems, structures and components important to safety. Procedural and configuration controls of items important to safety including validation of required labels are PFP restart prerequisites.

5. **In the case of a simultaneous failure of Dampers B and C, there is a possibility of the spread of contaminated material from Zone 3 to 1 from the fabrication area 235-C. How much contaminated material is stored in this area and what is the possibility of it escaping the glove box during this event?**

Response. Former weapons fabrication area Rooms 235-B and 235-C are really one large room. Glovebox HA-23S is a large storage glovebox that contains reactive scrap in plastic jars. Other gloveboxes have only transient inventories while work is being performed. Gloveboxes are kept at a negative pressure relative to the room by one of two methods. The dry air supplied gloveboxes such as those connected to Conveyor-Glovebox HA-28 in Rooms 235-B and 235-C are controlled at the point of supply and exhaust by diaphragm operated valves with large diaphragms having one side open to the glovebox and the other to the room. The large diaphragm size enables the slight differences in pressure to be the motive force for the valves, thus isolating them from instrument air failures. The other gloveboxes, such as 235-B-1, are connected to the E-4 exhaust which draw air through a HEPA filter from the room. These gloveboxes are always negative since the E-4 pressure is at about -4 in. water and the pressure upsets described in Section 9.1.7 do not exceed this figure.

6. For the full range of ventilation upset conditions, it is stated in Section 9.1.7A.1.3 that a fully trained operator could attempt to re-balance the system and continue. A less than fully trained person would trip the steam turbines and call for assistance. Do your procedures quantify the difference between a fully trained experienced operator, a fully trained in-experienced operator, and a less than fully trained operator? What would be the consequences if a trained in-experienced operator or a less than fully trained operator tried to re-balance the system?

Response. This statement was an evaluation of basic human nature rather than on adequacy of training. The kind of event that is described, inadvertent positioning of a control system valve, would be next to impossible to correctly diagnose or correct from the control room. As described an operator would not know this until he tried. An inexperienced person, having less self confidence would be more likely to take the "fall back position" of just dropping to turbines as is done in other hard to cope with situations. An experienced person with more self confidence might think that the situation was recoverable and try vainly to cope. In the end this person would also have to go to turbines but it would take longer to give up trying. As discussed in the response to General Safety Review Question 5, Westinghouse is doing an extensive review of implementation of the FSAR as part of the current facility restart program. This effort includes validation of systems, structures and components important to safety. Procedural and configuration controls of items important to safety including validation of required procedures and training are restart prerequisites.

7. **Comments on Loss of Ventilation analysis (Section 9.1.8):**

The analysis seems to assume that the fans stop instantly on loss of electric power; it seems more reasonable to assume that the fans will coast down over a period of several minutes due to impeller inertia.

Response: The purpose of the analysis is to show what happens after all fans have stopped. For years speculation has caused worry about reverse flow, both powered and from natural causes. We feel that our description of the powered ventilation system was adequate to dispel that part of the concern but we did not know the extent of the natural effects. Thus the analysis was performed to do that. As a consequence it became apparent that some of the ventilation equipment that had been considered to be essential was really not. Thus the coast-down period of the fans is really not germane to this analysis.

- Using the building volume figures given on Page 9-179 and assuming 400,000 cubic-foot for Building 291-Z and the 291-Z stack and using standard temperature and pressure air density of 0.075 lb/cubic-foot, we get total weight of air in the buildings of approximately 300,000 lb rather than the 252,000 lb cited on Page 9-175.

Response: Again, the analysis was not intended to be rigorous. The combined volume of Building 291-Z where ventilation air exists, as roughly calculated from prints, and of the stack comes to about 114,000 ft³ or about 3% of the total of buildings 234-5Z and 236-Z. Please also note that the inside air was assumed to be a conservative 80° F and not standard temperature.

- **It appears that the equation for mass of air on Page 9-175 should be $M = W/g$ rather than $M = W/2g$.**

Response: The 2 before the g was placed there incorrectly but was not included in the calculation. The 2 will be removed at the next update of the document.

- **The assertion on Page 9-178 that the flow resistance will vary as the square of the velocity is true for turbulent flow; for laminar flow the flow resistance will vary linearly with the velocity. As the flow coasts down, eventually it will become laminar. Using the turbulent relationship all the way to zero flow probably resulted in an overestimate of the coast down time; ignoring blower impeller inertia probably caused an underestimate of coast down time.**

Response: The reviewer is correct but again this was NOT intended to be a rigorous analysis.

- **All in all, the results reported on Pages 9-176 and 9-177 need to be recalculated. An HVAC code should be used with a simple nodalization of the ventilation system.**

Response: The value to be derived from the rigorous analysis does not appear to be worth the cost. Again, this study was only intended to give the roughest indication of magnitude and duration.

- **The calculation of Q on Page 9-180 assumes a complete building air change in one hour under loss of forced ventilation flow conditions. It appears the desired calculation should be:**

$$\begin{aligned} V_2 - V_1 &= (P_1/P_2) * V_1 - V_1 \\ &= [(P_1/P_2) - 1] * V_1 \\ &= [(P_1 - P_2)/P_2] * V_1 \\ &= [(29.21 - 29.06)/29.06] * 3,600,000 \text{ cubic-foot} \\ &= 18,600 \text{ cubic-foot} \end{aligned}$$

Assuming this volume change equilibrates during the hour that the pressure is changing, the flow rate will be:

18,600 cubic-foot/60 min = 310 cubic-foot/min

Response. The reviewer is correct but it shows that a reverse flow due to atmospheric changes is even less of a problem. Since the error does not result in an increase in risk the correction will be made at the next update.

Please discuss each of the above interpretations further.

Response. Please see the comments above.

- 8. Has a high energy line break been considered as an accident initiator, including breaks at any location that could create a direct or indirect challenge to one or more of the critical safety functions? For example, has the potential for a high energy line break to rupture the HVAC ducting been considered, including the potential for the steam release to cause inadvertent damper actuation such that an HVAC flow disturbance more severe than previously considered or evaluated may occur? Has jet impingement and or pipe whip been considered? Has the potential for the steam line supply to the HVAC turbines to rupture or leak and feed back into the HVAC exhaust ducting been considered, evaluated, and documented, if so, where? Has the potential for a steam leak in the supply air HVAC after conditioner been considered as a potential initiating event for some process or component malfunction? Has loss of steam supply and the potential effects on the supply air and resultant process disturbances been considered? Has the potential for these conditions to degrade ability of equipment to perform safety functions been evaluated? Where are these evaluations documented? Does this evaluation comprehensively model the HVAC system, including all potential failure/disturbance sources, and assess the overall contribution of HVAC faults to a radiological release?**

Response. High energy line breaks have been considered but the location of the lines (Section 5.4.6) and the nature of the "critical safety systems" (Table 4.4-2) are such that a break would not result in a problem. The highest energy lines in the PFP are 225 lb/in.² steam lines. These lines are routed through the second floor and the duct level. The general distribution is via pipes at 50 lb/ in.² or less. The next highest pressure gas is air which is distributed at 90 lb/in.² through the duct level.

A high energy break of a steam line that would rupture schedule 10 stainless steel E-4 ducting was not considered nor was rupture of the other lighter gauge supply and E-3 ducting considered. Such a break in passive systems would be quite unlikely during any time but during a seismic event when much of the duct work is assumed to break anyway.

Steam would not cause a worse damper actuation than is analyzed because we hypothesize complete opening or closing of the dampers.

Jet impingement has not been considered as the steam lines are not located near vital safety equipment.

The steam turbines and lines are exterior to the heavy gauge steel fan enclosures and a rupture could not enter the plenum.

Steam has leaked in the supply plenum with little effect. The air is humidified by water washes routinely and only part of the air supply comes through any given enclosure. Please refer again to the list of safety class equipment in Table 4.4-2.

Loss of steam is considered as an initiator in a ventilation upset in Section 9.1.7F.1.3.

Ventilation upsets are documented in Section 9.1.7 where the system is modeled and radioactive released assessed.

9. **Has the probability of plugging the inlet for the HVAC been determined and the potential consequences been analyzed? Has an HVAC system upset condition, during maintenance that uses a "greenhouse," been considered with the associated impact on the release potential? Please discuss the consequence of these scenarios.**

Response. The probability of plugging the inlet has not been determined but the potential has been analyzed in Section 9.1.7F.

The effect on a greenhouse is evaluated in Elements D-17 and D-18, Page 9-48. Greenhouses are normally ventilated with portable exhausters and balanced relative to the room. Thus, they are not sensitive to upsets of one room relative to the next room or to the outside. When panel changes are being made so the greenhouse is open to glovebox internals, Vent and Balance personnel are present and standing-by the glovebox exhaust controls to mitigate upsets.

10. **Has the potential release path from HVAC ducting rupture and through the building siding panels been considered? For example, a seismic event may cause some deformation to the building siding panels such that a direct release path to the atmosphere is established for the radioactive material contained in the failed ducting and disturbed filter media. From the existing seismic analysis this seems to be a viable scenario, but it is not specifically evaluated in the facility safety analysis report. Please discuss this scenario further.**

Response. The effect of duct work failure and building breach is presented on page 9-533 to 9-536 and 9-494 to 9-501.

11. **Are the glove box air supply HEPA containers restrained and sealed against a significant reversal of air pressure/flow and has this capability been tested?**

Response. Each 8 in. by 8 in. plywood framed HEPA filter is fastened by 4 approximately 1/4 in. diameter bolts. They are not tested as the frame would be much stronger than the media of the filter. The weak point in gloveboxes in regard pressurization are bag-out ports and the sphincter seals that are used to pass in small tool or other items in cardboard, one quart ice cream cartons. Experience has shown that these leak first.

12. **The evaluation of steam line break in a glove box and the resultant pressurization should consider the potential for the outlet damper to close as a result of the steam release or due to random failure, and the impact which would have on the resultant release from the glove box.**

Response. The only glovebox with a steam line is in the Engineering Laboratory. The exhaust damper for this glovebox is a hand operated damper in the duct level secured in place by a wing-nut. Excess flow would only tend to open the damper. The impact of a closed damper would be to over pressurize the glovebox as described in general in Section 9.1.9.

13. **After a criticality, the fission product gas concentration and release rates depend on an air exchange of 6 room changes per hour. What is the justification for the 6 air changes per hour and the size of the room? How will higher or lower air exchange rates affect release rates and off-site doses? Are individual room air exchange rates monitored? What is the justification for air exchange rates not being a limiting condition for operation?**

Response. Rooms in the PFP are ventilated at a rate of 6 air changes per hour without recirculation.

The small size of the room gives a conservative value for air concentration.

The release rate as related to room air concentration is not used in calculation of dose (Van Fleet, 1990a).

Room air rates are monitored from time to time.

Air exchange rates need not become a limiting condition of operation as they have negligible effect on the dose received by off-site individuals. The only reason that we included the calculation was to satisfy a requirement in Regulatory Guide 3.39, Section 9.2.1.2.3.b., for delineation of radionuclide concentrations in the confinement atmosphere following accidents.

**DP-31 comments regarding PFP FSAR chapters 4.0 and 9.0
revisions dated July 19, 1994
(August 4, 1994)**

1. **Page 9-38: Please explain how 0.33g of plutonium nitrate solution results in 0.16g of Pu.**

Response. The chemical formula for plutonium nitrate is $\text{Pu}(\text{NO}_3)_4$. Taking the molecular weight of Pu as 238, the molecular weight of a plutonium nitrate molecule is $238 + 4(14 + 3 + (16)) = 486$. Thus, Pu accounts for $238/486 * 100\% = 48.97\%$ of the plutonium nitrate mass.

2. **Page 9-251: Why is it acceptable for the estimated dose to an offsite receptor to meet the risk acceptance guidelines for an event frequency of $3.8 \times 10^{-2}/\text{yr}$? Based on historical data, the probability of a criticality occurring in the DOE complex is more likely to be approximately $10^{-1}/\text{yr}$ which was originally assumed in the PFP FSAR (1991).**

Response. The probability of a criticality at a DOE processing facility is shown to be $2.5 \times 10^{-2}/\text{yr}$ in section 9.2.3.4 of the SAR. Other than minor editorial changes, the content of this section is identical to the 1991 version. Since the acceptable dose increases as frequency decreases, meeting the acceptance criterion for $3.8 \times 10^{-2}/\text{yr}$ means that we are below the acceptance criteria for $2.5 \times 10^{-2}/\text{yr}$.

3. **Page 9-258: Please update the discussion regarding the seismic qualification of the 2736-Z complex to include WHC's most recent seismic analysis work.**

Response. This section has been updated.

4. **Page 9-354: States that "the method used in sections 9.2.1F and 9.2.4A.6.3.2.C to determine TNT equivalence of the explosion will be used here." The method is no longer presented in section 9.2.1F and does not appear to be presented in section 9.2.4A.6.3.2.C. Is this method appropriate? If it is an appropriate method please discuss the method used to determine the TNT equivalence on this page.**

Response. TNT equivalence method is not presented in the referenced sections and is not appropriate for the H_2 generation scenario in section 9.2.4A.6.3.2.A.n (p. 9-354 in the July 19, 1994 version). This section is being rewritten.

5. **Page 9-368: Please provide further discussion regarding the 2000g administrative limit on Pu handled in glovebox MT-5. Specifically, discuss the 2000g limit with**

respect to criticality safety regarding a seismically induced criticality (i.e. provide the basis for the 2000g limit so that readers will understand its significance).

Response. This section has been updated to include a discussion of the 2 kg administrative limit.

6. **Page 9-439: Section 9.2.4B.2.4 has been deleted. This section provided a discussion regarding a seismically induced criticality occurring in glovebox MT-5. Please provide justification for why this section no longer needs to be included. refer to the above comment)**

Response. The replacement section for 9.2.3B.2.4 was inadvertently left out of the July 19, 1994 version.

7. **Page 9-443. Please provide further justification for why the dose consequences resulting from both the "ventilation" and "no-ventilation" cases are acceptable. For example, why is it acceptable for doses to meet the risk acceptance guidelines for an event with a frequency of $1 \times 10^{-4}/\text{yr}$?**

Response. Per telephone conversation with PNL staff on August 8, 1994, sections 9.2.4.1.1, 9.2.4.1.2, and 9.2.4.1.3 will be rewritten to clearly state the calculated doses, the seismic event frequency, and the pertinent risk acceptance guidelines.

8. **Page 9-456, Last Sentence: The accident scenario hypothesizing the loss of 7000 gallons of HNO_3 in the presence of 90 mile per hour winds assumes '... that the aerosolization effects due to the high winds and the dynamics of the spill are small enough that the conservatism built into the evaporation analysis provides a sufficient margin of safety'. It is indeed likely that this would be conservative if all the inventory was assumed to aerosolize, but realistic wind stability classes were used (i.e. A or B), then the subsequent downwind airborne concentrations at the maximally exposed onsite and offsite receptors would be lower than the concentrations resulting from pure evaporation with a F stability wind field. Has an analysis been conducted with all liquid assumed to aerosolize instantly (or even more conservatively, a instantaneous release of equivalent mass of vapor) using A stability 19 m/s wind speed to support the assumption of conservatism in the model?**

Response. As discussed in various phone conversations, it is difficult to characterize the amount of aerosolization from a pool due to a 90 mph wind. The most applicable experiments that WHC is aware of are discussed in DOE-HDBL-0013-93, section 3.2.5.1. The referenced experiments were performed by Robyler and Owczarski [1] and measured the aerodynamic entrainment of salt aerosols from sea water as a function of wind speed and distance from the lee shore. Unfortunately, the highest wind speed considered was 15 m/s (33.6 mph). It is worth noting, however, that for a

pool the size in the HNO_3 spill (100 m^2), the release rate is on the order of $1 \times 10^{-7} \text{ kg/s}$ for a 15 m/s wind. Thus, using the dispersion wind speed of 19 m/s (which is not too far removed from the experimental data) and increasing the aerosolization rate by an order of magnitude to cover uncertainties due to extrapolation, we would expect to see a release rate on the order of $1 \times 10^{-6} \text{ kg/s}$. This would be $1 \times 10^{-6} / 4.6 \times 10^{-2} \times 100\% = 0.002\%$ of the evaporation rate. Based on the data presented in DOE-HDBK-0013-93, the aerosolization rate increases by approximately an order of magnitude when the wind speed is doubled. Thus, extrapolating the 15 m/s data, the release rate for a 40 m/s wind is on the order of 0.1% of the evaporation rate.

Per discussion with PNL staff, this section has been revised to consider a 10% aerosolization rate as the HNO_3 is released from the tank. Note that the previously reported 6 in transfer line is actually a 1 in line.

9. **Page 9-482, Last Sentence:** Section 9.2.8.3.3 describes the use of a model attributed to Britter and McQuaid for analysis of an initially dense gas release of CCl_4 . The analysis recognizes that the model is only applicable to C or D stability wind fields, in contrast to the actual scenario hypothesis of a F wind stability class. Presumably in an effort to compensate for the lack of generality of the model, the analysis assumes that no dispersion occurs between the point of release and the downwind distance at which the Britter and McQuaid model predicts that the dense gas cloud will transition to a neutrally buoyant plume. The implied assertion here is that the degree of conservatism introduced by assuming no dilution of the dense gas compensates for the lack of applicability of the Britter/McQuaid model in the case of F stability wind conditions. While this assertion may again be true, it is difficult to support conclusively without additional scoping calculations. It may be simpler to select one of the well known public domain dense gas codes (e.g. SLAB, DEGADIS, ADAM, etc ...) and perform the dense gas dispersion analysis in a more standard, rigorous fashion.

Response. Unfortunately, the dense gas dispersion code available at WHC is DEGADIS, and it has not been properly certified for use in a SAR. Once the need for dense gas dispersion calculations was identified, insufficient time remained to obtain certification for any of the usual dense gas codes. As a confirming calculation, however, WHC has run the ALOHA emergency response code. ALOHA is based on DEGADIS and was benchmarked by the developers of (NOAA) against DEGADIS "to ensure that only minor differences existed in results obtained from both models." The dense gas dispersion model in ALOHA was run with a direct source of 1.19 kg/s of CCl_4 as in Section 9.2.8 with the result that the onsite receptor located 550 m from the spill was subject to a maximum concentration of 48 ppm . This is a factor of three below the hand calculations presented in section 9.2.8.

DP-31 Comments regarding WHC-SD-CP-SAD-009
Seismic Accident Scenario for Current Conditions at 232-Z
November 29, 1994

1. **Section 5 -- Does the Langer respirable release fraction of $2.4E-04$ (refer to page 17) bound the release from the glovebox toppling event? The cited respirable release fraction (RRF) of $2.4E-04$ appears to be in error. Section 4.4.5.2 of DOE-HDBK-0013-93 gives an RRF of $3.3E-04$ for drops onto aluminum oxide on a pad.**

Response. The value of $3.2E-04$ is the correct value to use for this portion of the analysis. All calculational results that were derived from the erroneous $2.4E-04$ have been changed throughout the document. The conclusions of the document are not impacted by the slightly higher releases calculated using the correct value.

2. **The basis for concluding that the 232-Z Building will not be damaged by a 0.09g earthquake relies on two reports, one by EPRI and the second by Oak Ridge. The use of general conclusions from these reports is not adequate for qualifying seismic performance of the 232-Z Building. For example, some of the walls in Building 232-Z are nearly 16 ft. high. This may or may not be typical of the buildings studied in the two reports. The basis for performance of the structural components comprising the 232-Z Building during a seismic event need to be derived from structural calculations. It is our understanding that WHC has performed an analysis of the 232-Z building for a 0.20g pga. This document should be referenced in this safety document and used as the basis for establishing seismic performance. It is expected that a copy of the analysis will be forwarded to DOE (DP-311) for a cursory review and informational purposes. It is understood that the calculational package is not being used for seismic qualification of the 232-Z Building for the DBE, but only to establish a basis for concluding that Building 232-Z will not collapse below a 0.09g pga seismic event.**

Response. The results of the calculations contained in WHC-SD-WM-DCR-002, "232Z Building Seismic Assessment" have been included in the text of Section 7.1 to strengthen the risk acceptance discussion and the document has been included as a reference. A copy of the document has been provided directly to the DOE-HQ reviewers as previously agreed to.

3. **The SAD should be revised to provide a clear, concise description of the seismic performance of structures and equipment assumed in the determination of dose consequences. The SAD contains misleading statements concerning what seismic performance is assumed in the analyses (that which can be demonstrated) versus arguments about what is realistic (which, if included, should only be used to indicate the conservatism of the analyses). This is particularly a problem in**

Section 4.0; in Section 5.1, Glovebox Damage, Case 2, and Case 3; Section 5.2; Section 7.1; and Section 7.2.

Response. The document was revised throughout to clarify structural performance, the assumptions used in the analyses and the definition of the bounding scenario. In particular the last column of Table 5-2 was revised to clearly identify the Bounding Scenario. These changes are of an editorial nature. No changes to the technical elements of the document were made.

General Response: The comments discussed below were reconciled by incorporation into Appendix B of WHC-SD-CP-SAD-002 that will be issued as Revision 1. This Appendix is the entire text of WHC-SD-CP-SAD-009 Revision 0 modified only to reflect the responses to the DOE-HQ comments and minor editorial changes.