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7. Abstract

The purpose of this accident safety analysis is to document in detail, analyses whose results were reported in summary form in the *K Basins Safety Analysis Report* WHC-SD-SNF-SAR-001. The safety analysis addresses the potential for release of radioactive and non-radioactive hazardous material located in the K Basins and their supporting facilities. The safety analysis covers the hazards associated with normal K Basin fuel storage and handling operations, fuel encapsulation, sludge encapsulation, and canister clean-up and disposal.

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K BASINS SAFETY ANALYSIS

WESTINGHOUSE HANFORD COMPANY

DECEMBER 1994

**For the U.S. Department of Energy
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1.0 INTRODUCTION

The purpose of this accident safety analysis is to document in detail, analyses whose results were reported in summary form in the *K Basins Safety Analysis Report* WHC-SD-SNF-SAR-001. The safety analysis addresses the potential for release of radioactive and non-radioactive hazardous material located in the K Basins and their supporting facilities. The safety analysis covers the hazards associated with normal K Basin fuel storage and handling operations, fuel encapsulation, sludge encapsulation, and canister clean-up and disposal.

A hazard identification process was completed to systematically and thoroughly review the K Basins Fuel Encapsulation and Storage facility design and operations for hazards and select, based on severity and frequency (risk), accidents and abnormal operations for further review (Strickland 1994). A Preliminary Hazards Analysis (PHA) was developed to identify the K Basins Fuel Encapsulation and Storage Facility hazards. The PHA is a qualitative technique which identifies hazards and provides a basis for accident analysis.

The PHA was originally developed using Hazards and Operability (HAZOP) methodology and was issued in 1991 (Siemer and Peterson 1991). The HAZOP was reevaluated and updated by a team composed of personnel from K Basins Operations, Independent Safety and Safety Analysis and Engineering in 1994 (Strickland 1994) to account for the current facility configuration and to assure that existing safety analyses addressed all significant accident scenarios.

The HAZOP process systematically examined facilities and systems in the 100 K Area supporting irradiated fuel storage and encapsulation activities and included the following items.

- 181-KE river pump station,
- 183-KE water treatment plant,
- KE/KW clearwells,
- 190-KE service water pump station,
- 1706-KE demineralization plant (replaced),
- Recirculation and cooling systems for the 105-KE basin,
- Mixed resin bed ion exchange system and associated filters for basin water cleanup,
- 1908-K outfall to the river.

The Qualitative Accident Severity Levels, from WHC-CM-4-46, Appendix B, have been segmented into four severity categories (category I, II, III, IV). Each one has a definition of the consequences to the Public, Workers, or Environment. The consequences are ranked (in order of decreasing severity). Frequency Ranges found in Appendix B, contain the information regarding the likelihoods (frequencies) of the various initiating events. They are also ranked in order of decreasing likelihood. They are segmented into 4 probability categories (Anticipated, Unlikely, Extremely Unlikely, and Incredible) and each one has a category description and a frequency.

Criteria for selecting Hazards and Operability items for further study was based on consequences only. Only those items ranked as Category

I or Category II were considered significant. Category I consequences may cause deaths onsite or loss of the facility/operation, major injuries or illness offsite, radiation exposure to offsite individuals in excess of annual limits, or severe impact on the environment. Category II consequences may cause severe injury or severe occupational illness onsite, exposure to onsite individuals in excess of annual limits, major damage to a facility/operation, minor illness or injury offsite, exposure to offsite individuals to radiation below annual limits, or major impact on the environment. Note that in accordance with a recent revision to WHC-CM-4-46, the qualitative accident severity levels have been removed from Appendix B. However, at the time the PHA was conducted, and later when it was reviewed, Appendix B was active and was the major criteria for selecting PHA items for further analysis.

The majority of Category I and Category II items identified in the HAZOP were found to be addressed in the existing KE/KW Safety Analysis Report and Safety Evaluation Report. Modifications to the water chlorination system, installation of a new water demineralization system, and installation of an air chiller to cool basin water were examined. That examination identified five accident scenarios having possible Category II consequences that were not covered in the original HAZOP (Siemer and Peterson 1991). These scenarios were added to the list provided in the PHA (Strickland 1994). The accident scenarios have been analyzed in Chapter 8 of the SAR.

In a separate effort from updating the PHA, Safety Analysis and Engineering personnel evaluated the adequacy of existing K Basins safety documentation. A spray leak was analyzed as part of the Section 8.0 ion exchange column drop accident. Since the ion exchange columns concentrate radionuclides that were in the basin water, any other spray leak involving basin systems would be bounded by this accident.

For the accidents then, those possibilities from the PHA table that showed the highest potential risk or consequences were then chosen, and along with the existing scenarios to be analyzed, were documented in the following sections.

2.0 CRITICALITY SAFETY EVALUATION REPORT SUMMARIES

2.1 CRITICALITY SAFETY EVALUATION 105-KE BASIN FUEL ENCAPSULATION

BACKGROUND - N Reactor irradiated fuel is currently stored in the K East (KE) reactor storage basin in open storage canisters on the basin floor in regular constrained arrays. The fuel will be removed from the open canisters and placed in sealed Mark II canisters (encapsulated) to provide an additional confinement barrier. The encapsulation process introduces the potential for accumulation of broken fuel at the encapsulation station and canister handling errors beyond those for normal canister handling and storage, and accumulation of canisters that contain scrap fuel only.

ANALYSIS - An analysis has been made (Schwinkendorf, et. al, 1992) that identifies the criticality safety limits and controls required to safely encapsulate the Mark IV 0.947 wt% ^{235}U enriched fuel. The analysis also provides the technical basis in support of the limits and controls. The required criticality safety limits and controls for the encapsulation of the fuel is as follows:

Mass of uranium fuel and scrap in the Encapsulation Area	Maximum of 2,031 kg (equivalent to 6 canisters)
Maximum enrichment for material to be handled	0.947 wt% ^{235}U
Maximum amount of fuel assemblies or broken pieces in an open or closed canister (equivalent to fourteen 66 cm [26 in.] fuel assemblies)	328.4 kg (724.1 lb)

The mass limit of 2,031 kg (4,478 lbs) is equal to 42% of a critical mass in hemispherical geometry. By comparison, the $k_{\text{eff}} = 0.98$ mass (3,282 kg, 7,239 lbs) for optimum size scrap (0.5 in. O.D. rods) in hemispherical geometry is calculated to be 68% of a critical mass (4,793 kg, 10,567 lbs).

Under normal operating conditions, the k_{eff} for the facility would be in the 0.66 to 0.76 range. The "worst normal" configuration, which corresponds to the 2,031 kg (4,478 lbs) limit collected together into an optimum moderated and reflected hemispherical configuration, would have $k_{\text{eff}} = 0.89$ for intact Mark IV fuel, and $k_{\text{eff}} = 0.95$ for optimum-sized unirradiated scrap. Additional contingencies described in the next section would have to take place to reach or exceed a $k_{\text{eff}} = 0.98$. Note that these limits apply to unirradiated fuel which is more reactive than irradiated fuel.

According to the double-contingency criterion, no single accident should result in a criticality or in the framework of the analysis exceed a $k_{\text{eff}} = 0.98$.

- Three accounting errors (the first, second, and third contingencies) resulting in three extra canisters worth of fuel material in the encapsulation area would approach $k_{\text{eff}} = 0.98$. The assumption is

made that the three additional canisters contain no intact fuel elements.

- The weight of a canister is in error due to a scale calibration drift. The scale to be used has maximum error of 0.3% of rated capacity and 330 weighings would be necessary to accumulate a mass error equal to one canister. This is conservative, since random errors add quadratically, not linearly. Scale drift would have to be reset once every twenty work days to prevent scale errors from accumulating beyond one canister.
- Overloading of scrap piece canister is basically a mass control question and would be detected at weigh out. A canister containing 1.3 cm (0.5 in.) OD rods at optimum spacing would contain 251 kg (554 lbs) of scrap. Additional scrap would result in closer spacing, which is progressively less reactive. A canister containing 338.5 kg (746 lbs) of uranium rods is undermoderated, and any additional scrap will decrease reactivity further.
- Canister cleanup station broken fuel piece buildup needs to be monitored. Any periodic cleanup operations need to consider materials in this location to avoid buildup of material equivalent to a canister weight.
- Homogeneous sludge buildup is not a limiting criticality control issue. The computer codes as well as detailed experiments show that uranium oxide in water cannot be made critical at 0.947 wt% ²³⁵U enrichment.
- The packager has a small narrow opening that may allow fuel pieces to gain entry into an inaccessible location. To conservatively model this condition where fuel pieces are present, core cross sections for the optimum 1.3 cm (0.5 in.) OD rods to have a $k_{eff} = 0.6853$ will be used. Piling the 2,031 kg (4,478 lbs) hemisphere on top of the packager table resulted in a k_{eff} less than for the hemisphere water reflected on all sides. Since the entire volume could hold up to 3,043 kg (6,710 lbs) of optimally-sized close-packed fuel pieces, this amount of metal would be of concern during post encapsulation cleanup or whenever the packager would require lifting or moving. Provisions should be made to clean the volume accessible by the slot unless the size of the debris can be confined to fines that are not a criticality concern.
- Another contingency (analyzed in Wittekind 1991) was the possibility that water could be lost in the space surrounding filled canisters, leaving water inside. This would reduce the amount of neutronic isolation between canisters, but leaving them well moderated inside. The k_{eff} for this scenario increased from $k_{eff} = 0.6702$ (basin full of water) to $k_{eff} = 0.8902$ with the basin dry. This calculation was for an infinite array of double-stacked canisters containing Mark IV fuel. Many other scenarios were analyzed in (Wittekind 1992), including various heights of homogeneous and heterogeneous sludge and scrap surrounding canisters to varying depths. No scenario resulted in exceeding a $k_{eff} = 0.98$.

- The criticality safety analysis for storage of irradiated N Reactor fuel in the KE Basin was previously reported (Toffer and Eaves 1982). This reference establishes the safety of transferring canisters above the fuel storage array in the basin, by analyzing various accidents involving dropping fuel canisters amongst the stored canisters.

SUMMARY AND CONCLUSIONS - The mass control approach is a viable systematic method for implementing criticality control in the encapsulation area in the KE fuel storage basin during the encapsulation of N Reactor fuel and broken fuel pieces. The encapsulation process will place fuel and broken pieces into Mark II canisters via processing at the mass limit. The approach complies with double-contingency requirements and provides adequate margins for performing the planned operations safely. The critical mass limit is based on optimum size rods. This conservatism neglects the fact that the fuel is irradiated (irradiated fuel is not as reactive as unirradiated fuel) and that most of it is intact. Passive or engineered controls are not viable for encapsulation.

2.2 CRITICALITY SAFETY EVALUATION 105-KE BASIN FUEL CANISTERS IN SLUDGE

BACKGROUND - N Reactor irradiated fuel is currently stored in the K East (KE) reactor storage basin in canisters. There is an accumulation of material, designated as sludge, containing uranium oxide of variable composition and density around and possibly underneath these irradiated-fuel storage canisters. The material contains ill defined quantities of diluent. During the encapsulation process, emptying the canister will introduce additional uranium oxide sludge and parts of broken fuel elements too small to recover and be placed in broken fuel element canisters. Storage in K West (KW) basin is in sealed containers. Uranium oxide is not a component of the lesser amount of accumulating sludge present there so criticality concerns are not addressed.

ANALYSIS - An analysis has been made (Wittekind 1992) of the criticality considerations associated with this sludge containing broken fuel pieces and the associated fuel storage canisters. As a limiting case model, the sludge was modeled as wet uranium oxide of different densities between about 3 g/cm^3 and 6 g/cm^3 (.11 lbs/in³ and .22 lbs/in³) uranium oxide. Both MCNP and WIMS-E (Gubbins et. al. 1982, Halsall & Taubman 1986) computer code calculations were made for the case of 0.947 wt% ²³⁵U uranium oxide around the Mark IV fuel canisters with and without uranium rods to simulate broken fuel pieces in the uranium oxide sludge. Unirradiated uranium (no credit taken for ²³⁵U depletion) was used in the analysis as a conservative assumption. For most scenarios k_{eff} was computed for configurations containing stainless steel and aluminum canisters. These configurations represent the limiting case for reactivity increase due to sludge accumulation; calculations were also performed for a realistic sludge composition derived from analytical measurements.

Two types of calculations were performed for comparison to benchmark experiments - one involving Mark IA subcritical tests and the other involving homogenous mixtures of $\text{UO}_3 - \text{H}_2\text{O}$. These two benchmarks are considered closely related systems, and because adequate calculational techniques exist for relatively small extrapolation from these data, the k_{eff} limit of 0.98 is justified according to Company policy (WHC-CM-4-29).

SUMMARY AND CONCLUSIONS - There is nothing to indicate a potential criticality hazard involved with sludge accumulation around or storage of sludge canisters between the irradiated fuel canisters in the KE fuel storage basin. Specific results include:

- The sludge minimum slab thickness is 125.36 cm (49.35 in.) for k_{eff} of 0.98.
- Canisters containing Mark IV fuel with water moderation were more reactive than canisters containing uranium oxide (simulated computer model of sludge) and slightly less reactive than canisters containing uranium dioxide with uranium metal rods (simulated computer model of broken fuel in sludge).
- For uranium oxide accumulations to 90 cm (35.4 in) outside canisters containing Mark IV fuel or canisters containing fuel rods in uranium oxide) the MCNP calculated k_{eff} remained below 0.98 for both stainless steel and aluminum canisters.
- Canisters containing uranium oxide or canisters containing uranium rods imbedded in uranium oxide stored between rows of Mark IV fuel canisters does not result in significant reactivity increases even for the case where the canisters are surrounded by uranium oxide.

2.3 CRITICALITY SAFETY EVALUATION OF THE 100-K BASIN ION EXCHANGE MODULES AND ION EXCHANGE COLUMNS

Criticality Safety Calculations (Erickson 1994a) have been made to provide the technical basis for establishing limits associated with operation and storage of the K Basin Ion Exchange Modules (IXMs) and Ion Exchange Columns (IXCs).

BACKGROUND - The 100-K Basin IXMs and IXCs are used to remove radionuclides from the basin water. The basin water flows through a filter, passes through the IXMs/IXCs, and flows back to the basin. An IXM consists of six steel cells with resin containers set in concrete. After depletion the IXMs are buried. The IXCs are individual steel resin columns installed in cells. After depletion, the IXCs are temporarily stored into either a concrete burial box or a lead vault. For final disposition, they are sent to the solid waste group for burial.

IXMs are changed out every six months or earlier unless analysis shows they have not accumulated more than 110 grams (3.85 ounces) of plutonium, and are still efficiently removing contaminants. IXCs are changed out every one to two months as needed according to analysis.

Under normal circumstances, the IXM or IXC is depleted with inorganic cations and anions consisting of Ca^{++} , Mg^{++} , Na^{+} , Cl^{-} , SO_4^{--} and HCO_3^{-} ions as well as various radionuclides present in the basin water. The radionuclides compete with the other ions for the active sites on the ion exchange media. The IXMs and IXCs will preferentially remove the ions with higher valences before the radionuclides with lower valences. Most of the uranium and plutonium in the basin water is in a neutral valence state, and therefore the IXMs and IXCs will not be retaining either metal due to ionic interaction, but will be acting as a mechanical filter. Using uranium in

the form UO_2^{++} , which is a very unlikely valence state, yields a resin saturation limit of 0.243 g/cm^3 ($.009 \text{ lbs/in}^3$) of uranium.

Normal Operation Analysis - For an IXM at the saturation limit of 0.243 g/cm^3 ($.009 \text{ lbs/in}^3$) and assuming that the uranium to plutonium ratio is 400:1 (conservative estimate from basin water analysis) a k_{eff} of 0.111 is calculated for 0.95 wt% uranium and a k_{eff} of 0.149 is calculated for 1.25 wt% uranium. For an IXC at the limit of 0.243 g/cm^3 ($.009 \text{ lbs/in}^3$) and assuming a U:Pu ratio of 400:1, a k_{eff} of 0.109 is calculated. Using the uranium inlet concentration of 2.5 E-4 g/l and the removal efficiency of 95%, it would take about 2.4 years to accumulate this material. When the IXC is drained for shipping, the calculated k_{eff} is 0.116. With 1.25 wt% enriched uranium in place of the 0.95 wt% enriched uranium, considering KW Basin operations, resulted in a k_{eff} of 0.139.

Off-Normal Condition Analysis - The analysis used the following conservative assumptions: the IXM or IXC is left in service until saturated with ions, and the only ions available in the basin are uranium or plutonium (or a mixture of the two). Three off-normal condition scenarios were considered. For the first case, the IXMs and IXCs were homogeneously filled, all of the material was plutonium (uranium and other materials were not considered) and there was no physical limit placed on how much plutonium could be held. For the second case, the IXMs and IXCs only had the top one-third filled, all of the material was plutonium (uranium and other materials were not considered) and there was no physical limit placed on how much plutonium could be held. The final case had a uranium to plutonium ratio of 10:1 instead of the 400:1 for normal conditions, and there was no changeout time limit.

With all IXM cells filled homogeneously, and using the assumed IXM inlet concentration of 1.5 E-06 g plutonium per liter (based on sample analysis at the KE-Basin) it would take about 63 years to accumulate sufficient plutonium to reach a $k_{\text{eff}} = 0.95$. If only the top one-third of the IXM cell is used, it would take about 30 years to accumulate sufficient plutonium to reach a $k_{\text{eff}} = 0.95$. Assuming a normal six month operating time, this represents at least 60 times the normal operational lifetime of an IXM. For the final case with the uranium to plutonium ratio of 10:1 with no H_2O and an IXM limit of 0.243 g/cm^3 for fissile material, the $k_{\text{eff}} \approx 0.75$ is calculated. The absence of H_2O corresponds to a drained IXM. This case is safely subcritical.

For the IXCs and using the same assumptions as before, it would take about 25 years to accumulate plutonium to reach a calculated k_{eff} of 0.95. If only the top one-third of the IXC is used It would take about nine years to accumulate plutonium to reach a $k_{\text{eff}} = 0.95$. Assuming a normal two month operating lifetime, this represents at least 54 times the normal operational lifetime of an IXC. For the final case using the uranium to plutonium ratio of 10:1 with H_2O and the IXC limit of 0.243 g/cm^3 ($.009 \text{ lbs/in}^3$) for fissile material, a $k_{\text{eff}} = 0.717$ is calculated. If the IXC is drained then k_{eff} is 0.673. If 1.25 wt% enriched uranium is used then $k_{\text{eff}} = 0.715$ with water and $k_{\text{eff}} = 0.675$ without; the results are all safely subcritical.

For the IXCs in Concrete Burial Box, the IXCs are assumed to be in a close packed square 5 by 5 array. Each IXC would need to have been in operation for about 13.7 years for plutonium to accumulate to reach a k_{eff}

= 0.95. The case with the uranium to plutonium ratio of 10:1 and the IXC limit of 0.243 g/cm^3 ($.009 \text{ lbs/in}^3$) for fissile material, the $k_{\text{eff}} = 0.834$ with water and $k_{\text{eff}} = 0.914$ without. For 1.25 wt% enriched uranium, $k_{\text{eff}} = 0.842$ with water and $k_{\text{eff}} = 0.924$ without. These cases are all safely subcritical.

A parametric study of interspersed moderation in the concrete burial box was also completed. The results show that the most reactive configuration is with no interspersed moderation present, and the addition of interspersed moderation lowers the calculated k_{eff} .

SUMMARY AND CONCLUSIONS - A criticality in an IXM or an IXC while in use, or while a IXC is stored in the lead caves or the concrete burial box is not credible. The quantities of uranium and plutonium needed for a criticality and the time necessary to accumulate those quantities would not allow a criticality. Most uranium and plutonium is actually held up by the filtering action of the IXM. Therefore, the quantities used in the calculations based on ionic trapping are extremes and cannot be obtained. All analyses ignored the presence and buildup of non-fissionable materials in the ion exchange materials. In the off-normal conditions the IXMs are still safely subcritical with double contingencies considered. The different contingencies are leaving the IXMs online for many times the normal time combined with only accumulating plutonium, or accumulating both uranium and plutonium at a ratio of 10:1 which is much more plutonium than the normal 400:1.

Based of the criteria of the Nuclear Criticality Safety Manual, WHC-CM-4-29, a CAS is not required.

2.4 CRITICALITY SAFETY EVALUATION FOR K BASINS SANDFILTERS

BACKGROUND - The K Basins sandfilters are used to filter contaminants from the basin water coolant. Coolant water is skimmed from the surface of the basin and pumped through the sandfilters. Particulates in the water from different sources, including corroded fuel, are mechanically captured in the sandfilter matrix. As the particulates accumulate, the pressure drop across the sandfilter increases, requiring periodic backwashing to remove these particulates and permit continued operation. The backwash water is pumped into the sandfilter backwash pit. The criticality of the sandfilter backwash pit is covered in a separate report (Erickson 1994b).

ANALYSIS - An analysis/evaluation has been made to verify operation of K Basins sandfilters does not require the presence of an operational criticality alarm system (CAS).

The sandfilter k_{eff} 's were calculated for three normal or off-normals:

- Addition of Material - homogeneously distributed in sandfilter medium
- Moderation - sandfilter dryout with fixed fissile inventory
- Uranium, Plutonium Heterogeneity - thin layer on sand filter media top

The sandfilter reactivity was calculated for five parametric studies:

- Plutonium Segregation - $\text{PuO}_2 + \text{H}_2\text{O}$ distributed in sandfilter medium
- Moderation - sandfilter dryout with fixed fissile inventory
- Plutonium Heterogeneity - thin layer on sand filter media top
- Loss of Neutron Absorbers - 0.947 wt% ^{235}U uranium considered
- Loss of Neutron Absorbers - 1.250 wt% ^{235}U uranium considered

The analysis was based on conservative assumptions regarding material composition and quantity, i.e. 100 percent removal efficiency from the coolant for an operating period far exceeding the planned life of the facility versus the normal practice of back flushing every three to six months. The analyzed composition of the sand in the sandfilter was combined with the measured uranium and plutonium isotopic composition in the criticality calculations. The sandfilter contains other constituents including fission products, activation products, and corrosion products from the stored spent fuel and other materials in the basin for which no credit was taken. The presence of these other constituents in the analysis would provide an additional safety margin. In addition to ignoring the presence of these other materials, parametric studies considered plutonium separated from uranium, and unexposed uranium, both 0.947 and 1.250 wt% ^{235}U , in the sandfilter medium.

SUMMARY AND CONCLUSIONS - The analysis shows based on assumptions associated with normal operating conditions and off-normal operating conditions, and based on other very conservative assumptions, that a criticality ($k_{\text{eff}} > 1.00$) is not credible. For off-normal conditions, fissile material accumulations, reduced uranium to plutonium ratio, and change of moderation due to dryout with fixed fissile inventory were considered.

Regarding the need for a Criticality Alarm System (CAS), based on the criteria of WHC-CM-4-29, the Nuclear Criticality Safety Manual, no CAS is necessary associated with sand filter operation. Criticality analysis has determined that no realistic situation could result in a criticality due to the physical form and isotopic distribution of fissionable material.

2.5 CRITICALITY SAFETY EVALUATION OF THE 100 KE BASIN SANDFILTER BACKWASH PIT

BACKGROUND - The KE Basin sandfilter backwash pit is used to hold the backwash from the basin coolant cleanup system sandfilter. Coolant skimmed from the surface of the basin is pumped through the sandfilter. The sandfilter is regularly backwashed to remove the material that has built up in it and pumped into the backwash pit. The particulates in the backwash pit water tend to settle to the bottom of the pit and form sludge.

ANALYSIS - Criticality Safety Calculations (Erickson 1994b) have been made to provide the technical basis for establishing a safe mass limit for the sludge for continued operations in the KE Basin sandfilter backwash pit. The analysis in this report is based on a uranium-plutonium-water system using the measured isotopic concentrations for the uranium and plutonium. The actual sandfilter backwash pit sludge contains many other constituents including fission products, activation products, and corrosion products from the stored spent fuel and other materials in the basin. The presence of these other constituents in the analysis would provide an additional significant safety margin. In addition to ignoring the presence

of other materials further conservative assumptions include: geometry of the sludge, hydrogen to uranium ratio (H:U), and full water reflection during off-normal conditions. The analysis is based on the 0.94 wt% enriched uranium contained in KE Basin and would not be valid for KW Basin backwash pit sludge that could contain some 1.25 wt% enriched uranium. However, because the fuel elements in the KW Basin are encapsulated, there is relatively less sludge in the backwash pit.

SUMMARY AND CONCLUSIONS - This analysis shows that under normal operating conditions a criticality ($k_{eff} \geq 1.00$) is not possible even with the backwash pit completely full of sludge. For off-normal conditions the minimum hemispherical and spherical safe mass of plutonium for the sandfilter backwash pit over a wide range of U:Pu ratios are given. These data presented in the analysis can be used to establish a limit for the quantity of plutonium allowed in the sandfilter backwash pit. The data used to generate the plutonium safe mass curves are very conservative. The analysis considers a number of contingencies (geometry, moderation, material addition, and uranium-plutonium segregation). The analysis ignores the presence of neutron absorbers known to be present in the sludge.

With the addition of the fission products and other materials found from chemical analyses of sludge from the sandfilter backwash pit, at actual densities and concentrations, the sandfilter backwash pit is safely subcritical under all conditions of moderation even for infinite quantities of sludge.

2.6 CRITICALITY SAFETY EVALUATION OF THE 100 K BASIN CARTRIDGE FILTERS

BACKGROUND - A nuclear criticality analysis was performed for filter cartridge cage assemblies used in the Hanford Site KE and KW Basins. The previous administrative limit was to prevent plutonium buildup in each filter cartridge assembly beyond 225 grams. Each filter cartridge assembly holds 88 filter cartridges arranged in a cluster configuration. The filter cartridge cage assembly removes particulates from the basin water by trapping materials in the polypropylene fiber of the filter cartridges.

Normal operation was analyzed (Schwinkendorf 1994b) by performing explicit calculations of the filter cartridge assembly cluster geometry using the W-PIJ module of the British WIMS-E code. Data were obtained for an average uranium and plutonium concentration typical of the center-of-basin in KE Basin. These data, along with the flow rate through the filter cartridge assembly, allowed for the estimation of the fissionable material buildup inside the filter assembly with time. Several off-normal contingencies were analyzed, including buildup of fissionable material beyond the planned replacement period, loss of geometry control within the filter assembly, and lower-than-expected U/Pu ratios. The effects of interspersed moderation were also calculated, which would occur as the filter cartridge assembly is raised out of the basin during replacement and placed in dry storage.

ANALYSIS - The filter cartridge cage assembly contains 88 filter cartridges placed in concentric rings. Each filter cartridge is 30 inches long, and is made of a polypropylene fiber wound around a hollow 304 stainless steel core. The stainless steel core wall is 0.011 inch thick,

and has an outer diameter equal to 1.0646 inch. The outer diameter of the filter cartridge is 2.5 inch. The filter cartridges are held in place by two vertical plates, pulled together with tie rods. The diameter of the filter cartridge assembly is equal to 30 inches. Filter cartridge cage assemblies have a water flow capacity of 28,500 gallons per hour (107,873 ℓ/hr). Two cartridge assemblies are installed to insure that one is available for service at all times. The filter cartridge cage assemblies are located near the bottom of the basin floor in the K Basins. The two filter cartridge cage assemblies are separated by a sufficient amount of water to provide neutronic isolation, approximately 20 inches.

Discussion of Contingencies--Off-normal scenarios considered in this criticality safety evaluation include failure to replace the filter cartridge assembly when required to do so (resulting in an ever increasing loading of fissionable material) and the loss of geometry in the filter cartridge assembly (resulting in optimal spacing of the filter cartridges within the assembly). Also considered were lower-than-expected values for U/Pu and the loss of fission and corrosion products.

The maximum amount of sludge material to accumulate inside the filter cartridge medium considered credible was a "thick paste" of 4.0 g/cm^3 heavy metal oxide solution, where the heavy metal was composed of both 0.95 wt% ^{235}U enriched uranium, and 12 wt% ^{240}Pu plutonium. The U/Pu ratio was assumed to equal 400 for normal conditions, and U/Pu = 10 was used in contingency analysis; U/Pu = 10 is lower than any U/Pu data available for KE Basin, and so this is considered as a bounding case. Extensive chemical analysis of K Basin sludge samples has established that U/Pu = 400 is a best-estimate. This value is also consistent with reactor production calculations. The nominal time required for this amount of buildup (4.0 g/cm^3 of mixed uranium plutonium oxide in the available void spaces inside the filter medium) was determined to be just over 2 years at a U/Pu = 400.

As fissionable material builds up inside the filter cartridge assembly, the amount of water moderator inside required for maximum reactivity will shift to maintain the optimum. At the maximum credible plutonium loading analyzed (4.0 g/cm^3 of metal oxide in solution) the system is undermoderated with full density moderation. At intermediate values of fissionable material loading, optimum moderation would be expected to take on intermediate values.

Normal Conditions--The filter cartridge is a particulate removal system very similar to the sand filter. It will remove particles of uranium oxide, plutonium oxide, fission products, and corrosion products. These materials will be trapped by the polypropylene fiber and accumulate there. In the sand filter the sand was periodically cleaned by a back wash. The particulates collected by the sand filter were washed into the sand filter backwash pit. Extensive characterization of the sludge has taken place, detailed chemical and isotopic analyses have been performed. The sludge composition is well known. The filter cartridge should be removing the same type of sludge from the water. Therefore as the filter cartridge becomes loaded it will have very similar sludge in it as found in the backwash pit.

Under normal operation the filter cartridges will collect sludge consisting of fission and corrosion products and uranium and plutonium as

oxides in a ratio of $U/Pu = 400$. Calculations (Erickson 1994b) have shown that at these U/Pu ratios with fission products and corrosion products present, the filter cartridge cage can not be made critical under any conditions irrespective of whether the uranium enrichment is 0.95 or 1.25 wt% ^{235}U .

If a very conservative assumption is made that the fission products and corrosion products disappear and we are left only with a uranium and plutonium oxide system in the filter cartridge, the system still can not be made critical with any amount of plutonium present. In other words the k_{eff} will be less than 0.95 for typical conditions of moderation and material loadings. The highest k_{eff} calculated for a 10 year accumulation of uranium and plutonium is equal to 0.813. This conclusion is true both for KE and KW Basins. Therefore for normal operation of filter cartridges in water and in dry storage will be substantially subcritical and will not exceed $k_{eff} = 0.95$.

The first calculations for normal conditions assume that uranium and plutonium loading builds up in the filter void spaces until the time the filter is replaced. The "void fraction", or volume fraction inside the filter cartridge medium that is available for water intrusion and material uptake, can be estimated by subtracting the smear density to solid density ratio from one, or $(1 - 0.2445/.9) = 0.7283$.

This void fraction was assumed to be filled with a sludge material composed of 4.0 g/cm^3 metal oxide solution. This metal oxide solution consists of UO_2 plus PuO_2 , according to a given U/Pu ratio. A U/Pu equal to 400 was considered for normal operation. This is thought to represent a maximum amount of fissionable material that can be considered as credible inside the filter cartridges; this "solution" would actually be a thick paste.

Furthermore the calculations assumed that there are no fission products, iron, or other corrosion products present in the system. This conservative assumption resulted in significantly higher calculated neutron multiplication than would otherwise be the case.

An average value for uranium and plutonium g/L appropriate for the center-of-basin and the flow rate through the cartridge filter assembly (28,500 gal/hr) were used to calculate the accumulation rate of uranium. Given this rate, it is possible to find (assuming 100% filter efficiency) the time required for this maximum amount of material to accumulate. Assuming that $U/Pu = 400$, the time required for this density of 4.0 g/cm^3 heavy metal oxide sludge to accumulate is approximately 2.1 years. The plutonium is held constant when the U/Pu is changed in the model because higher confidence is given to the measured plutonium data than to the measured uranium data. In the next section on contingency analysis, $U/Pu = 10$ is considered. The time required to fill the filter medium with 4.0 g/cm^3 material is approximately 40 times longer (there is 40 times less heavy metal in solution). The results in Table 2-1 are based on an infinite array of filter cartridge assemblies (an albedo boundary condition of 1.00 was imposed on the outer surface of a 20 cm water reflector around the filter cartridge assembly), but only a single layer high; an axial leakage correction appropriate for a cylinder 30 inches high was used. At high U/Pu ratios the neutron multiplication factor shows some sensitivity to uranium enrichment. To model a single cage assembly, the middle two rows in Table 2-1 were redone using an outer boundary albedo equal to zero.

With 20 cm of water reflector, the infinite-array system is essentially decoupled. These results are in the last two rows in Table 2-1. There are two entries in the last column of Table 2-1. The first entry is for both moderator and reflector density equal to 0.01 g/cm^3 . This models the filter cartridge assembly after it is initially pulled out of the basin. The second entry is for moderator density equal to 0.01 g/cm^3 , but 20 cm of full density water reflection is placed around the filter assembly. This models the filter cartridge assembly in dry storage, but fully reflected.

Table 2-1. WIMS-E W-PIJ k_{eff} Results for Maximum Sludge Accumulation.
(U/Pu = 400, time = 2.1 years without replacement)

^{235}U enrichment	albedo	Full water density	Water density = 0.01 g/cm^3
1.25	1.0	0.813028	0.213715 / 0.226095
0.95	1.0	0.735925	0.193586 / 0.205021
0.95	0.0	0.726024	0.192760 / 0.198250

The results in Table 2-1 clearly show that when the best-estimate U/Pu value of 400 is assumed, there is no danger from a criticality, even if the filter cartridge assembly is not replaced after two years of continuous operation, which would be a gross violation of procedures. Even assuming that the uranium enrichment is equal to 1.25 wt% (which would not be the case in the KE Basin), the k_{eff} remains well below 0.90. It is also apparent from the second entry in the last column that full density reflection around the drained filter cartridge assembly does provide some additional moderation to boost the k_{eff} somewhat, but not to a significant degree. This effect will also be shown in the next section, where this table is repeated for the contingency where U/Pu = 10.

The results in Table 2-1 are appropriate for filter cartridges that have not been replaced in two years. Reduced accumulation calculations were performed to find k_{eff} as a function of time before replacement. As fissionable material builds up, it makes sense that the amount of water moderation required for an optimum reactivity configuration would also shift, and so interspersed moderation calculations were performed as a function of time before replacement. All of these time-dependent calculations assumed a boundary albedo equal to zero. Figure 2-1 illustrates the k_{eff} versus time for full density as well as several reduced-density cases. The U/Pu ratio for these results was equal to 400.

For the U/Pu = 400 scenario, there is so little fissile material in the filter cartridge assembly that full density moderation is not optimum. It is difficult to see in Figure 2-1 that this is the case because the variation in k_{eff} due to moderator variation is small compared to the scale of the graph. However, Figure 2-2 illustrates the variation in k_{eff} as a function of interspersed moderator density for two points in time, which correspond to metal oxide concentrations equal to 0.5 and 1.0 g/cm^3 in the available void space in the filter cartridge medium. The highest value shown in Figure 2-1 corresponds to the bottom row in Table 2-1, for U/Pu =

Figure 2-1. k-effective vs Time.

U/Pu = 400, U-235 = 0.95 wt%

Figure 2-2. k-eff vs Water Density.

U/Pu = 400, U-235 = 0.95 wt%

400, where $k_{eff} = 0.726$. These curves would increase somewhat for an infinite array of filter cartridge assemblies, up to $k_{eff} = 0.736$ for an infinite array (hexagonal lattice spacing equal to 44 inches center-to-center). If the assumed uranium enrichment is increased to 1.25 wt%, this maximum would increase to 0.813, which is still substantially below the k_{eff} safety limit of 0.95, and this would require over 2 years of continuous operation.

Normal operation is considered to be continuous operation out to the scheduled replacement time, which is six months for KE Basin. Assuming that the U/Pu is equal to 400, Figure 2-1 shows that the k_{eff} of a single cartridge assembly, fully water reflected, is less than 0.34 when replaced on schedule at the end of six months of continuous operation.

Off-Normal Conditions--For off-normal conditions the following contingency sequences can be considered. The first contingency is the assumption that there are no fission or corrosion products in the filter cartridges. The second contingency can be assumed to be inadvertent material buildup, changes in the U/Pu ratio, collapse of the filter cartridge assembly geometry, or change in moderation.

Inadvertent material buildup without a change in the U/Pu ratio will not exceed $k_{eff} = 0.95$ for collapse of geometry or loss of moderation. If a material buildup occurs (because basin personnel neglect to replace the filter cartridge assembly on schedule) with a concurrent loss of uranium (i.e., the U/Pu ratio is equal to 10), the k_{eff} will approach 0.95 after 10 years, where the plutonium is building up at 520 g/yr in the filter cartridge assembly.

The first off-normal contingency analyzed was failure to replace the filter cartridge at the scheduled time interval of six months. If the U/Pu ratio remains at 400, extended operation will increase k_{eff} significantly, as shown in Figure 2-1, but even after 2 years of continuous operation, the continued accumulation of fissionable material never drives the k_{eff} of the filter cartridge assembly beyond 0.80. Double contingency does not result in a k_{eff} greater than 0.95.

The second off-normal contingency analyzed was for the U/Pu ratio to be less than the best-estimate value of 400. There are measured data that show a large spread in U/Pu values, but little confidence is given to this data because of the level of uncertainty in the measurements. Nevertheless a value of U/Pu = 10 was selected as a bounding case for the uranium loss scenario. Table 2-1 was redone for U/Pu = 10 and the results are reported in Table 2-2. At this U/Pu ratio and typical plutonium concentration in the basin water, filter cartridge assemblies would go critical before the 78 year (4.0 g/cm^3 in filter body) accumulation time assumed in Table 2-2. These calculations were also performed as a function of time. Figure 2-3 shows k_{eff} as a function of time for the case where U/Pu = 10. The highest reactivity is achieved with full density moderation. Because of the steep rate of increase in k_{eff} with time, increased resolution was given to the time out to 1.5 years, shown with an expanded scale in Figure 2-4. Each curve in Figure 2-4 is plotted for optimum moderation. For each curve, the optimum moderator density increases as fissionable material accumulates; each point represents the maximum for that amount of accumulated material (as interspersed moderator density was varied). Figure 2-3 shows that even

Figure 2-3. k-effective vs Time.

U/Pu = 10, U-235 = 0.95 wt%

Figure 2-4. k-effective vs Time.

Optimal Moderation with Time.

with a U/Pu = 10, it would take nearly twelve years of operation before a criticality could take place, which is nearly 24 times the scheduled replacement interval. The results in Figure 2-4 show that with U/Pu = 400, k_{eff} is less than 0.36 at the time of replacement, and k_{eff} is less than 0.54 at one year of operation. The additional curves in Figure 2-4 illustrate behavior for intermediate values of U/Pu. The curves for both U/Pu equal to 200 and U/Pu = 400 terminate where the heavy metal oxide has reached 4.0 g/cm³. The rate of rise for U/Pu = 400 is steeper because total heavy metal is accumulating faster (plutonium concentration is held constant).

However, the higher plutonium content of the U/Pu = 200 heavy metal oxide mixture results in a higher k_{eff} when 4.0 g/cm³ is finally reached, although this occurs later in time. The final point plotted for U/Pu = 100 is when the metal oxide content is equal to 3.0 g/cm³. The final points plotted for the U/Pu = 10 and U/Pu = 25 curves are for metal oxide contents equal to 0.5 g/cm³ and 1.0 g/cm³ heavy metal oxide, respectively.

Table 2-2. WIMS-E W-PIJ k_{eff} Results for Maximum Sludge Accumulation.
(U/Pu = 10, time = 78 years without replacement)

²³⁵ U enrichment g/cm ³	albedo	Full water density	Water density = 0.01
1.25	1.0	1.232345	0.365224 / 0.379434
0.95	1.0	1.229298	0.363638 / 0.377833
0.95	0.0	1.214307	0.363321 / 0.368335

It takes more than two contingencies to reach a $k_{eff} = 0.95$.

The third contingency considered was if the filter cartridge assembly were to come apart. It is conceivable that the tie rods could break during the replacement procedure, in which case all geometry control would be lost. In this scenario, optimum spacing must be assumed. In Figure 2-5, the lattice k_{∞} of filter cartridges is plotted as a function of time, out to 9.7 years without replacement. The U/Pu ratio is the most conservative, U/Pu = 10, and full density moderation was assumed. This would be the case if the filter assembly were to come apart while still under water (i.e., such as while being raised out of the basin during replacement). Figure 2-5 shows that if the filter cartridge assembly comes apart while still under water, $k_{\infty} = 0.95$ will not be exceeded as long as this occurs before four years of operation. The k_{∞} at 3.9 years (equivalent to 0.2 g/cm³ heavy metal oxide sludge in the void spaces inside the filter medium) is equal to 0.938 when the filter cartridges are brought into a close-packed arrangement (hexagonal touching). Further accumulation of fissionable material could then produce a k_{∞} greater than 0.95 when brought into optimum spacing, but the finite amount of material would be correspondingly large, depending on how close k_{∞} is to $k_{eff} = 0.95$. The point is that for the k_{∞} of the close-packed lattice to exceed 0.95 shortly after four years, the U/Pu must equal 10 and loss of geometry control must occur. These two contingencies (the cartridge filter assembly breaking apart and a value of U/Pu less than any measured) could occur and k_{∞} is still below 0.95. If the filter cartridge is not replaced on schedule and is left in operation

Figure 2-5. k_{∞} vs Spacing.

U/Pu Atomic Ratio = 10.

for over four years (which would be an additional contingency), then k_{∞} could exceed 0.95 for the compacted lattice. Additional fissionable material buildup beyond four years would be required for k_{∞} to exceed 0.95

Figure 2-5 illustrates k_{∞} values as fissionable material accumulates. The minimum spacing shown corresponds to touching in a close-packed hexagonal array. Beyond four years, the k_{∞} value may exceed 1.0, but the k_{eff} for a finite array will be smaller. Two-group lattice cross sections were extracted from the WIMS-E results and input to a neutron diffusion code, GOLF (Schwinkendorf 1994a), which calculated k_{eff} for a fully-water-reflected hemisphere of a size which gave either $k_{\text{eff}} = 0.95$ or $k_{\text{eff}} = 1.0$. The lattice spacing for the GOLF calculations were all for the most reactive, touching in a hexagonal lattice.

Figure 2-6 below shows the number of filter cartridges that would be required to attain a given k_{eff} as fissionable material buildup continues without replacement, if they were to come together into an optimum configuration in a fully-water-reflected hemispherical pile. It would take 6.2 years to gain sufficient reactivity for a single filter cartridge assembly (containing 88 individual filters) to come apart and have the individual filter cartridges rearrange themselves into an optimum lattice geometry and form a fully reflected hemispherical pile with a k_{eff} equal to 0.95. It would take over seven years of operation for this pile of 88 filter cartridges to attain $k_{\text{eff}} = 1.00$.

More than two contingencies have to occur to reach a $k_{\text{eff}} = 0.95$. In all the scenarios analyzed the double contingency was not violated.

Figure 2-6. Number of Filters.
Hemispherical Geometry, U/Pu = 10.

CONCLUSION - The Nuclear Criticality Safety Manual, WHC-CM-4-29, Section 5, states that:

"Criticality alarm systems (CAS) and criticality detection systems shall be installed as follows:

- (1) In those cases where the mass of fissionable material exceeds the limits established in paragraph 4.2.1 of ANSI/ANS-8.3 and the probability of criticality is greater than 1×10^{-6} per year (as documented in a DOE approved SAR), a CAS meeting ANSI/ANS-8.3 shall be provided to cover occupied areas in which the expected dose exceeds 12 rads in free air, where a CAS is defined to include a criticality accident detection device and a personnel evacuation alarm.
- (2) In those cases where the mass of fissionable material exceeds the limits established in paragraph 4.2.1 of ANSI/ANS-8.3 and the probability of criticality is greater than 1×10^{-6} per year, (as documented in a DOE-approved SAR), but there are no occupied areas in which the expected dose exceeds 12 rads in free air, a criticality detection system shall be provided; where a criticality detection system is defined to be an appropriate criticality accident detection device but without an immediate evacuation alarm. The criticality accident detection system response time should be sufficient to allow for appropriate process-related mitigation and recovery actions. While an immediate evacuation alarm is not required under these circumstances, evacuation shall be implemented (i.e., evacuation

notification or delayed alarm) if potential doses to occupational workers could be effectively limited by such actions.

- (3) In those cases where the mass of fissionable material exceeds the limits established in paragraph 4.2.1 of ANSI/ANS-8.3, but a criticality accident is determined to be impossible due to the physical form of the fissionable material, or the probability of occurrence is determined to be less than 1×10^{-6} per year (as documented in a DOE-approved SAR), neither a CAS nor a criticality detection system is required. In addition, neither a CAS nor a criticality detection system is required to be installed underwater when fissionable material is handled or stored beneath water shielding that is adequate to protect personnel; however, a means to detect fission product gases or other volatile fission products should be provided in occupied areas immediately adjacent to such underwater storage areas, except for fuel systems where no fission products are likely to be released. Also, neither a CAS nor a criticality detection system are required for fissionable material during shipment of fissionable material packaged in approved shipping containers, or fissionable material packaged in approved shipping containers awaiting transport provided no other operation involving fissionable material not so packaged is permitted on the dock or in the shipment area.
- (4) The decision to install a criticality detection system rather than a CAS, and the decision that neither a CAS nor a criticality detection system is necessary, must be justified based upon a documented DOE-approved safety analysis."

The basis for not requiring a criticality alarm system for the filter cartridge operation is that nuclear criticality is not credible during the operation and dry storage of the filter cartridges. The basis is founded on "due to the physical form of fissionable material."

The filter cartridge cage assembly is a filter for particulate matter in basin water, similar to the sand filter. The particulates that will be trapped include corrosion and fission products, uranium and plutonium oxides, and other particulates. The presence of corrosion and fission products in the filter cartridge assembly during normal operation in water and subsequent dry storage will assure subcriticality by a very significant margin.

For off-normal analyses, a number of contingencies were evaluated. In all accident scenarios, more than two contingencies were required to exceed the $k_{eff} = 0.95$ limit. Two of the more important contingencies were loss of all fission and corrosion products from the filter and the reduction in the U/Pu ratio. Even for multiple contingencies, for the filter cartridge assembly to reach $k_{eff} = 0.95$ continuous operation without replacement is required beyond the expected life of the facility, which is another 8 years.

3.0 CRANE FAILURE AND CASK DROPPED INTO LOADOUT PIT

3.1 SUMMARY

Crane failure leading to a cask drop into a K Basin will cause the basin floor slab to fracture. The resultant water flow rate through the fracture was estimated at 1,957 liters per hour (lph) [517 gallons per hour (gph)] with some uncertainties. At this leak rate water loss from the basin could be replaced by make-up systems. The calculated annual load drop frequencies lie in the "unlikely" range and no radiological dose consequences to onsite workers would result.

3.2 SCENARIO DESCRIPTION

A fully loaded transfer cask weighing 21,772 kilograms (48,000 lbs) is being transferred to the fuel loadout pit as described in Section 5.9 of the SAR. The crane used in the transfer fails and the load is dropped onto the floor of the transfer pit through a vertical distance of 7.6 meters (25 ft). The impact fractures the pit floor slab (Davis 1975), which is 21-inch thick, reinforced concrete.

The crane failure and load drop would be detected by the crane operator. An initial slight drop in the basin water level would be detected by facility operators.

3.3 FACILITY SPECIFIC BARRIERS

The walls and floor of the basin are the primary containment barriers and the underbasin membrane collects water from water leaks that may occur. Water leaks would flow by gravity to a sump where a pump circulates the water back to the facility. Two 303-lpm (80-gpm) vertical, extended-shaft-centrifugal sump pumps have been installed to return the leakage water collected in the sump back to the cooling pool. The pumps are equipped with electrical controls to alternate the pumping cycle from one pump to the other and operate both pumps when the demand exceeds the capacity of one pump. The sump pumps are not Safety Class 1 equipment.

3.4 ACCIDENT PHENOMENOLOGY

Table 3 of (Khaleel 1994) reports leak rates from a 16-m by 1.3-cm (52-ft by 1/2-inch) open construction joint in K Basin, based on 21 field-based measurements from K Area soils, as follows: 128 lph (33.8 gph) minimum, 15,182 lph (4,011 gph) maximum, 2,990 lph (790 gph) mean, 1,045 lph (276 gph) median, and 128 lph (33.8 gph) mode. There are large uncertainties in these results probably because of the small sample size. The mode flow rate (128 lph [33.8 gph]) is quoted by Khaleel as being the most likely flow rate. Use of the maximum flow rate was judged to be excessively conservative and not realistic. The mean flow rate (2,990 lph [790 gph]) was judged to be more realistic yet conservative.

When the dropped cask hits the water, some deceleration occurs due to water buoyancy but there is enough impact energy to shear a 1.6 m² (17 ft²)

section of the concrete basin floor and compress the soil under it. Davis 1975, estimates a compression depth of .23 m (22.7 cm) [0.744 ft (8.93 inches)] in the 53-cm (21-inch) thick floor slab. The assumption is made that a water flow path 2.54-cm (1-inch) wide around the perimeter of the cask bottom (1.2 m wide by 1.3 m long [4 ft wide by 4.25 ft long]) is created by the impact. The area available for flow is therefore $(1 \text{ in})([1 \text{ ft}/12 \text{ in}](2)(4 \text{ ft} + 4.5 \text{ ft})) = 1.42 \text{ ft}^2$.

By using the ratio of the area available for flow through the cask drop hole (.13 m² [1.42 ft²]) to the area available for flow through the construction joint (.2 m² [2.17 ft²]), the water flow rate through the cask drop hole is calculated as follows:

$$\left(790 \frac{\text{gal}}{\text{hr}}\right) \left(\frac{1.42 \text{ ft}^2}{2.17 \text{ ft}^2}\right) = 517 \frac{\text{gal}}{\text{hr}}$$

At this leak rate, water loss from the basin could be replaced by make-up water systems.

The probability of a crane failure is estimated from data in NUREG-0612 (George 1980). Crane load drop frequencies are found in this reference to range from 2.5 E-05/lift to 3 E-04/lift.

It is not known at this time how many crane lifts per year are expected; therefore, the following reasoning and assumptions are used: Fourteen fuel elements are housed in each fuel storage canister. At present there are 3,368 canisters stored in KE Basin and 3,821 canisters stored in KW Basin. Shipments from/to the basins are done by loading/unloading cask railroad cars. A cask railroad car holds three canisters for a total of 42 fuel elements. Assume that the contents of KE Basin will be shipped out over a period of two years. The number of lifts per year is therefore

$$\frac{3,368 \text{ KE canisters}/2 \text{ yr}}{3 \text{ canisters/lift}} = 562 \text{ KE lifts/yr}$$

Also, assume that the probability of a cask being over the basin water when the crane fails is 0.5. Therefore, on a yearly basis the crane load drop frequencies at KE Basin would range from

$$(2.5 \times 10^{-5} / \text{lift}) \left(562 \frac{\text{lifts}}{\text{yr}}\right) (0.5) = 7.0 \times 10^{-3} / \text{yr}$$

to

$$(3 \times 10^{-4} / \text{lift}) \left(562 \frac{\text{lifts}}{\text{yr}}\right) (0.5) = 8.4 \times 10^{-2} / \text{yr}.$$

This event frequency falls within the range of 10⁻²/yr to 10⁻⁴/yr, which is considered "unlikely" in accordance with WHC-CM-4-46 guidelines.

3.5 CONCLUSIONS

Crane failure leading to a cask drop into a K Basin will cause the basin floor slab to fracture. The resultant water flow rate through the fracture was estimated at 1,957 liters per hour (lph) [517 gallons per hour (gph)] with some uncertainties. At this leak rate water loss from the basin could be replaced by make-up systems. The calculated annual load drop frequencies lie in the "unlikely" range and no radiological dose consequences to onsite workers would result.

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4.0 DESIGN BASIS EARTHQUAKE

The Hanford Site is in a region of low to moderate seismicity. The historical record of seismic activity within a 322 kilometer (200 mile) radius of the 100-K Area site dates from 1827. During this time there have been only 14 earthquakes which may have been felt or were recorded at or near the 100-K Area site. Five of these may have affected the site with an intensity of IV or greater on the Modified Mercalli Scale (MM). The most severe of these occurred on December 14, 1872 with a probable epicenter located about 137-161 kilometers (85-100 miles) north of the facility. No historical record of the effect of the 1872 quake on the 100-K site is available, but it is postulated to be in the intensity range of V to VI MM. For more information concerning the seismology and geology of this area, see WHC-SD-GN-DB-003 (Tallman 1989), *Natural Phenomena Hazards: The Hanford Site*.

At the time the basins were constructed, the Uniform Building Code (UBC) was used to determine the design criteria for the basin structure. The current Hanford Site seismic design criteria for a non-reactor safety class 1 structure specifies a DBE defined as a maximum horizontal ground acceleration of 0.20 g simultaneously with a vertical ground acceleration of 0.13 g. The current design criteria is defined in Hanford Plant Standard SDC 4.1 (DOE-RL 1989). All current seismic analysis of the basins is based on the current safety class 1 DBE of 0.20 g.

4.1 BASIN STRUCTURE

The most recent dynamic structural analyses (Winkel 1991, Schmidt 1991) of the basin response to a 0.20 g DBE indicate that the basins structures will likely remain intact with only local cracking at the tapered walls in the K West Basin. However, the construction joint between the basin structure and the reactor building was determined to be vulnerable to failure during a postulated DBE. The construction joint between the basin and the reactor consists of an unreinforced vertical concrete joint with a flexible elastomer seal which runs the width of the basin. The dynamic analyses recently performed indicate the seal could be subjected to high stresses due to the interaction between the reactor building and the basin. The specifics concerning the failure of the flexible seal cannot be predicted. However, estimates of the leak rate resulting from a failed construction joint seal range from 18,925 lph (5000 gph) up to 94,625 lph (25,000 gph). These leak rates are in excess of the existing water makeup system capability and in excess of the currently analyzed basin leak accidents in this document.

A plan (Wiborg 1994) has been developed that calls for the installation of two isolation barriers (Figure 4-1) between the basin proper and the reactor building. Installation of the isolation barriers will isolate the main part of the basin where the spent fuel is stored from the reactor discharge chute where the construction joint is located (Figure 4-2). The isolation barriers are designated as safety class 1 structures and are

Figure 4-1. Main Basin With Isolation Dams Installed.

Figure 4-2. Discharge Chute With Isolation Dams Installed.

Figure Top view showing isolation of the discharge chute which is the location of the construction joint from the main basin. Isolation at this location would limit leakage, in the event an earthquake opens the construction joint. The water level in the main basin would remain at normal levels, assuring continued coverage of irradiated fuel in the basin.

designed and will be built to withstand the non-reactor safety class 1 DBE of 0.20 g horizontal and 0.13 g vertical ground acceleration. Dynamic structural analysis is currently ongoing to show the isolation doors will survive the DBE.

Following installation of the isolation doors, a seismic event may cause the construction joint between the basin and the reactor building to fail. However, even though the discharge chute area may drain, the fuel in the main basin will remain covered and cool, preventing any potential oxidation reactions or airborne suspension of radioactive material.

The plan will assess the impact of a leak from the basin into the soil under the basin. There is potential for a smaller sustained leak to develop even with the isolation doors installed. Make up water would be supplied to keep a sufficient water depth over the fuel for shielding as well as to keep the fuel cool (See Section 3.0). Also, the potential for release of radioactive material due to heatup of the fuel and possible oxidation reactions if the basin water were to drain completely will be assessed (See Section 5.0). The potential for the basin to drain completely following installation of the isolation doors is considered to be beyond the design basis for the basin.

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5.0 HYPOTHETICAL LOSS OF BASIN WATER ACCIDENT ANALYSIS

5.1 SUMMARY

Hypothetically, one of the most serious accident situations would be complete loss of water in the basin. However, it has been concluded that loss of water from the basin sufficient to uncover the fuel is not credible. The basin structure was analyzed for its ability to withstand the DBE. Also, the adjoining cask transfer pits were analyzed for a postulated cask drop accident. In both cases, the loss of coolant from the pool would be minimal, less than 25 gpm (see Sections 3.0 and 4.0).

However, an analysis have been made of the heat up of the basin fuel due to a hypothetical complete loss of basin water. This study (Beaver 1994) is based on a three dimensional representation of the storage basin, interior and surrounding structure and an array of 30 individual fuel storage canisters. This study determined that the maximum fuel temperature of intact fuel in closed containers is 181 °C (358 °F) for K-West Basin and 184 °C (363 °F) for K-East Basin. At this temperature the fuel cladding will remain intact. In addition, due to the large mass and low heat generation rate, the building and fuel heatup rate is sufficiently slow (approximately 1.3 °C/day [2.4 °F/day]) that operators would have days to ameliorate such an accident (Beaver 1994).

Although the fuel cladding remains intact during the hypothetical loss of basin water, the existing sludge on the bottom of the basin plus the corrosion products of fuel assemblies that were damaged upon removal from N Reactor still provides a potential source for the release of radionuclides. This analysis makes a conservative estimate of the radiological consequences of the aerodynamic entrainment of the sludge and fuel corrosion products in K-East basin as a part of the hypothetical loss of basin water analysis. The potential dose to the onsite worker was calculated to 0.12 rem EDE. The offsite dose was calculated to be 0.043 rem EDE.

5.2 SCENARIO DESCRIPTION

Starting Conditions and Assumptions--An unmitigated leak of 25 gpm from the basin is postulated for the K East basin. The basin water level would slowly decrease, the fuel canisters would uncover and eventually the sludge would begin to dry out. The particulate from the dry sludge would be aerodynamic entrained, carried outside the building and transported to the site boundaries.

The description of the hypothetical loss of basin water accident begins with a listing of assumptions used for this section followed by the accident sequence.

Various assumptions are made about the system to model the accident scenario. These assumptions are as follows.

- It is assumed that an unmitigated leak results in the total drainage of the pool.
- Releases of radioactive materials can occur in the form of solids, liquids, or airborne releases. Since airborne contamination is

the most prevalent health and safety risk, the accident analysis focuses primarily on airborne releases.

Development of the accident sequence was based on these assumptions.

Initiating Event and Sequence of Events--The accident scenario for the hypothetical loss of basin water assumes that an unmitigated leak of 25 gpm from the basin would cause the basin water level to slowly decrease. The fuel canisters would uncover and eventually the sludge would begin to dry out.

Detection--In the event that the basin water level dropped significantly, the operators would be aware of the situation over a long period of time. In addition, such a drop in the water level could result in unshielding of the basin bathtub ring and would cause a high radiation alarm requiring immediate evacuation of personnel.

Certain monitoring instrumentation is considered critical to the operation of the K Basins. One of the parameters monitored by this instrumentation is basin water level. The monitoring instrumentation has centralized indication and out of limit annunciation provided in the KE and KW control rooms. Trouble alarms exist in each control room to indicate possible problems from the other basin. The characteristics of the storage of fuel in the basins, however, does not require instrumentation with fast response or extreme accuracy.

Although basin water level changes could occur more quickly than changes in temperature or pH, most events that would cause rapid changes are evident to operating personnel, such as earthquakes or cask drops. Slow changes in basin level in the low direction would result in other effects occurring, such as detection of airborne radiation or increases in area radiation rates which would alert personnel to a problem being present. Procedures are provided for appropriate alarm response and corrective action. A backup water makeup valve is immediately available to the operating crews and can supply 1500 gpm to the cooling pool, to recover from reduction in water level. This makeup system is not Safety Class 1 equipment.

5.3 FACILITY SPECIFIC BARRIERS

The 105-K Basins, are enclosed within a building that while not serving as a containment would confine or mitigate any releases occurring at the basins.

5.4 ACCIDENT PHENOMENOLOGY

Release Due to Aerodynamic Entrainment of Sludge--An accident release quantity determination was performed to establish the amount of contamination released from the facility (the released source term).

Information and assumptions used to determine the amount of material released are as follows:

- In the event of an accident where the hypothetical loss of basin water occurs, it is assumed that an unmitigated leak from the

basin would cause the basin water level to slowly decrease. The fuel canisters would uncover and eventually the sludge would begin to dry out.

- Quantities of radionuclides present in the sludge and sludge density were obtained from the Attachment.
- Based on recent measurements of the depth of sludge in K-East basin, the volume of sludge is estimated to be between 2047 to 2147 ft³ (57 to 61 m³). Since moist sludge should retain particles due to the large surface tension force, no entrainment of particles from wet sludge should occur for achievable wind speeds at the Hanford site (Epstein and Plys 1994). Only the dry sludge material near the sludge surface will be vulnerable for entrainment. Particles as large as about 1.0 mm may be suspended in the boundary layer that flows along the surface of the sludge. Owing to their weight, most of the larger particles are re-deposited into the sludge surface were they impact and free smaller surface particles. It is assumed that only sludge material within a depth of ten times the diameter of these larger particles will be at risk to entrainment. This is equivalent to a depth of about 1.0 cm of sludge and includes 12.7% of the total sludge volume. Further supporting information is contained in the following paragraph.
- The bounding release fraction for aerodynamic entrainment for a homogeneous bed of powder under static conditions within a structure is $4 \times 10^{-6} \text{ hr}^{-1}$ (Mishima 1993) and the release is assumed to occur over an 8 hour period.

The mass rate at which sludge material is being entrained can be estimated from the sludge density and information already presented.

$$\begin{aligned}
 m'_{\text{sludge}} (\text{g/hr}) &= V (\text{m}^3) \times \rho (\text{g/cm}^3) \times RF (\text{hr}^{-1}) \\
 &= 47 \frac{\text{g}}{\text{hr}}
 \end{aligned}$$

The wind-induced particulate release from a drained K-basin has also been calculated from experimental data for particle entrainment coupled with a potential flow solution to estimate the air velocity induced at the bottom of the dry basin (Epstein and Plys 1994). A relationship for particle entrainment rate as a function of particle size and density, wind speed, surface density of particles and basin dimensions is developed in the report. This relationship is applied for 10 micron particles closely packed on the surface of dried sludge. For an average Hanford wind speed of about 8 mph, the particle source rate is 0.75 g/hr. For high wind speed of 45 mph, the source rate is 120 g/hr.

The 47 g/hr estimated using the aerodynamic entrainment release fractions of Mishima (Mishima 1993) falls with the range predicted by Epstein and Plys (1994). However, a more appropriate way to compare the two source term estimates would be to evaluate the corresponding air particle concentrations X , at the individual receptor locations.

For a release fraction independent of the wind velocity, all wind velocities are assumed initially to have the same quantity of entrained material. This allows a single X/Q' (units of g/m^3 divided by g/sec) value to be calculated. It can be seen that X/Q' is the ratio of air concentration and release. Multiplying X/Q' by the actual source rate gives an estimate of the particle concentration $X(\text{g/m}^3)$ at the receptor location which can then be converted into dose rate.

High wind velocities entrain more material, but, are less stable, more turbulent and disperse the material over a wider area. Although the lower wind velocities will entrain less material, their greater stability may allow them to transport the material more effectively to a particular location. For each wind velocity there is a corresponding dispersion factor and source rate. The atmospheric dispersion computer code GXQ Version 3.2 was used to calculate the air concentration, X using a velocity dependent source rate (Hey 1993). This updated version of the GXQ code is in the process of being validated and verified, and is used here only to show the relative conservative nature of the dose consequence calculations. Calculated air concentrations are shown in table 5-1 based on the sludge suspension rate developed in Epstein and Plys 1994. The results are two to three orders of magnitude less than those values based on a single release fraction from Mishima 1993. The larger Mishima X values demonstrate the conservatism in the release fraction used in the dose calculation.

Table 5-1. Comparison of Downwind Air Concentrations

Receptor Location	Air Concentrations, X (mg/m^3)	
	Mishima	Epstein and Plys
100 m radius (100 m E)	1.6×10^{-1}	2.2×10^{-3}
Near River Bank (480 m NW)	7.2×10^{-3}	2.4×10^{-5}
Fire Station (4000 m ESE)	9.6×10^{-4}	5.9×10^{-6}
1994 Hanford Site Boundary (12,040 m W)	3.4×10^{-4}	8.1×10^{-8}

The released source term is that quantity of contamination released from the K Basin during the accident sequence. The development of the source term that would be caused by a complete loss of water and entrainment of sludge from the basin is as follows.

The inventory at risk was determined by assuming that only the dry sludge material near the sludge surface will be vulnerable to entrainment. It was assumed that only sludge material within a depth of ten times the diameter of the larger particles will be at risk to entrainment. This is equivalent to a depth of about 1.0 cm of sludge and includes 12.7% of the total sludge volume.

Then, modeling a dried out surface of sludge, the total volume of sludge is multiplied by the fraction of the total that comprises the dry crust. This yields the inventory at risk.

$$\begin{aligned}\text{Inventory at Risk} &= 2150 \text{ ft}^3 \times 28,317 \frac{\text{cm}^3}{\text{ft}^3} \times 0.127 \\ &= 7.73 \times 10^6 \text{ cm}^3 \text{ sludge}\end{aligned}$$

The source term is calculated by multiplying the inventory at risk by the release fraction. Since the sludge was modeled as a homogeneous bed of powder, the release fraction of $4.0 \times 10^{-6}/\text{hr}$ was used. The release is assumed to occur over an 8 hr period. Thus, the released source term becomes:

$$\begin{aligned}\text{Released Source Term} &= 4.0 \times 10^{-6}/\text{hr} \times 8 \text{ hr} \times 7.73 \times 10^6 \text{ ml} \\ &= 247.36 \text{ ml sludge}\end{aligned}$$

Release From Damaged Fuel--An accident-release quantity determination was performed to establish the amount of contamination released from the facility (the released source term).

A survey of damaged fuel in the K basins is currently being performed, but is not complete. A preliminary estimate indicates that possibly 50% of the fuel has some form of damage sustained in the discharge from N Reactor. Corrosion of the metallic uranium has extensively degraded approximately 1 to 2 % of the fuel.

Information and assumptions used to determine the amount of material released are as follows.

- In the event of a complete loss of basin water and entrainment of corroded fuel from damaged fuel elements, it is assumed that each fuel element is broken once and two corroded fuel surfaces are exposed to the environment.
- For the modeling of a corroded surface, it is assumed that there is a "depth of corrosion" which when compared to the fuel element uranium mass and length will determine the fuel inventory at risk. Each broken exposed face of a fuel element is assumed to have a depth of corrosion of 0.001 meter from which radiological material can be released.
- For the method used here to model the corroded fuel, an assumption of 10% damaged fuel will yield a value of corroded fuel at risk that is consistent with the preliminary damaged fuel survey.
- Quantities of fuel and radionuclides present were obtained from the Attachment.
- The bounding release fraction for aerodynamic entrainment for a homogeneous bed of powder under static conditions within a structure is $4 \times 10^{-6} \text{ hr}^{-1}$ (Mishima 1993) and the release is assumed to occur over an 8 hour period.

The released source term is that quantity of contamination released from the K Basin during the accident sequence. The development of the source term that would be caused by a complete loss of basin water and entrainment of corroded fuel from damaged fuel elements is as follows.

From Table 2-6 of the Attachment, the fuel inventory in K East Basin is:

$$\text{K East Basin Total Fuel} = 1,146.16 \text{ MT fuel}$$

The inventory at risk was determined by assuming that the exposed surface area of the fuel was the predominant factor involved in the release of the radioactive material. The estimate of the amount of fuel at risk assumed that each fuel element was broken in one place. For each fuel element this gives two exposed fuel surfaces each having a depth of Δt . The fraction of the fuel in an element that is L long with two exposed fuel surfaces each having a depth of Δt , that is at risk for release is:

$$\begin{aligned} \text{Fraction at Risk} &= \frac{2 \pi R^2 \Delta t}{\pi R^2 L} \\ &= \frac{2 \Delta t}{L} \end{aligned}$$

Thus, the fraction of the fuel in an element that is 26.1 in long with two exposed fuel surfaces each having a depth of 0.001 m or of the seven elements in a canister that is at risk for release is:

$$\begin{aligned} \text{Fraction at Risk} &= \frac{2 \text{ surfaces} \times 1.0 \times 10^{-3} \text{ m/surface}}{26.1 \text{ in} \times \frac{2.54 \times 10^{-2} \text{ m}}{\text{in}}} \\ &= 3.0169 \times 10^{-3} \end{aligned}$$

Then, modeling a corroded surface, the fraction of fuel is multiplied by the total material inventory. This yields the inventory at risk.

$$\begin{aligned} \text{Inventory at Risk} &= 0.1 \times (1146.16 \times 10^3 \text{ Kg}) \times 3.017 \times 10^{-3} \times 10^3 \text{ g/Kg} \\ &= 3.46 \times 10^5 \text{ g fuel} \end{aligned}$$

The source term is calculated by multiplying the inventory at risk by the release fraction. Since the exposed fuel was modeled as a corroded surface, the release fraction of $4.0 \times 10^{-6}/\text{hr}$ for a corroded surface was used. Thus, the released source term becomes:

$$\begin{aligned} \text{Released Source Term} &= 4.0 \times 10^{-6}/\text{hr} \times 3.46 \times 10^5 \text{ g fuel} \times 8 \text{ hr} \\ &= 11.04 \text{ g fuel} \end{aligned}$$

Consequences Due to Aerodynamic Entrainment--Using the released source term, the radiological doses to the onsite and offsite receptors were determined. Information and assumptions used to determine radiological consequences to human receptors are:

- The maximum offsite individual is located 12.0 km west of the release point. For the maximum offsite individual only short-term exposure effects were evaluated. For the short-term exposure effects case, the pathways that were analyzed included inhalation and submersion. Offsite ingestion is not included because in the case of an accident, any contaminated land or products would not be used. Therefore, ingestion would not, be an actual exposure

pathway. The maximum onsite individuals are located 4000 m east south east of the release point at the Hanford fire station (Refer to the Appendix). The maximum onsite exposure was assumed to be exposed via both submersion and inhalation. A 50-yr commitment period for the time the ingested material remains in the body was assumed.

- From the attachment, the values of X/Q' for the selected receptor locations are presented in Table 5-2.
- The attachment recommends a value of 1.0×10^4 rem EDE/ml for as-settled sludge. This recommended value encompasses the range of variation in the sludge samples as well as incorporating both the EDE and limiting organ calculations.
- The attachment recommends a value of 2.3×10^5 rem EDE/g for fuel. This recommended value incorporates both the EDE and limiting organ calculations.

Using the dose consequences for the unit release generated by GENII (Attachment), the consequences resulting from a hypothetical loss of water from the basin and the entrainment of sludge and corroded fuel are calculated.

The atmospheric dispersion coefficients and unit doses provided in the Attachment can be used to determine the effective dose equivalent. This is done using the following equation from the Attachment.

$$D \text{ (rem)} = Q \text{ (ml)} \times \frac{X}{Q'} \text{ (s/m}^3\text{)} \times R \text{ (m}^3\text{/s)} \times C \text{ (rem/ml)}$$

where:

- Q = Quantity respirable radioactive material released (ml or g)
- X/Q' = Time integrated atmospheric dispersion coefficient
(Obtained from the Attachment Table 3-3)
- R = 3.3×10^{-4} m³/s standard man acute breathing rate (ICRP 1975)
- C = Dose per unit respirable radioactive material inhaled
(Obtained from the Attachment Table 2-16)

Consequences Due to Aerodynamic Entrainment of Sludge--Using this equation, the calculated EDE from sludge entrainment at the 1994 hanford boundary is 0.021 rem. Dose consequences calculated in the same manner for other selected receptors are given in the Table 5-2.

$$\begin{aligned} D &= 247.4 \text{ ml} \times 2.6 \times 10^{-5} \frac{\text{s}}{\text{m}^3} \times 3.3 \times 10^{-4} \frac{\text{m}^3}{\text{s}} \times 1.0 \times 10^4 \frac{\text{rem}}{\text{ml}} \\ &= 2.10 \times 10^{-2} \text{ rem EDE} \end{aligned}$$

Consequences Due to Entrainment of Corroded Fuel--Using the equation, the calculated EDE from the fuel corrosion entrainment at the 1994 hanford boundary is 0.022 rem. Dose consequences calculated in the same manner for other selected receptors are given in the Table 5-2.

$$D = 11.04 \text{ g} \times 2.6 \times 10^{-5} \frac{\text{s}}{\text{m}^3} \times 3.3 \times 10^{-4} \frac{\text{m}^3}{\text{s}} \times 2.3 \times 10^5 \frac{\text{rem}}{\text{g}}$$

$$= 2.18 \times 10^{-2} \text{ rem EDE}$$

Table 5-2. - Calculated Dose Consequences

Receptor Location	X/Q' (s/m ³)	EDE Sludge (rem)	EDE Fuel (rem)	EDE Total (rem)
100 m radius (100 m E)	1.2×10^{-2}	9.80	10.06	19.86
Near River Bank (480 m NW)	5.6×10^{-4}	0.46	0.47	0.93
Fire Station (4000 m ESE)	7.4×10^{-5}	0.060	0.062	0.12
1994 Hanford Site Boundary (12,040 m W)	2.6×10^{-5}	0.021	0.022	0.043

5.5 CONCLUSIONS

For the hypothetical loss of basin water, it has been shown that the fuel does not reach sufficient temperature to burn (Beaver 1994). Furthermore, dose consequence of the aerodynamic entrainment of the dry sludge for the onsite receptor at the Fire Station is 0.06 rem EDE. The offsite location at the present boundary is 0.02 rem EDE. Dose consequence of the entrainment of the both sludge and fuel for the onsite receptor at the Fire Station is 0.12 rem EDE. The offsite location at the present boundary is 0.043 rem EDE.

6.0 COMBUSTION OF URANIUM FUEL FOLLOWING DRYOUT

A potential energetic reaction may occur when canisters of fuel are opened in air or when the stored fuel at the bottom of the basin is exposed as the result of a loss of water from the basin. Failed fuel exposed to basin water oxidizes readily and uranium hydride may also be formed during the process of oxidation. Uranium hydride in a form as finely divided powder can be pyrophoric. If it were in this form and came in contact with air, it could in turn ignite the uranium. Uranium in fines burns readily, but has an elevated ignition temperature as a solid metal. Under the worst cases from all of the above conditions, the potential energetic reaction could release radioactive material from the basin causing high airborne activity and personnel contamination.

6.1 SUMMARY

This section presents the results of an evaluation (Epstein et al. 1994) of the thermal response of fuel elements stored in the Hanford K-basin following a hypothetical loss of basin water due to seismic damage. Key features of the evaluation are consideration of natural convection heat removal from the fuel, a critical review of experimental data regarding the ignition of metal fuel, and application of chemical reaction rate laws and ignition theory to determine the potential for fuel ignition.

Consistent with the heat transfer analysis, ignition theory demonstrates that natural convection into an open canister containing intact but decladded fuel elements results in stable (non-igniting) fuel temperatures. The fraction of hypothetical fuel hydride that results in the ignition of intact elements is quantified in Epstein et. al. 1994. The greatest and perhaps only threat to a stable drained K Basins argument is the hypothesized reactive hydride mechanism of fuel ignition. It is shown in Epstein et. al. 1994 that unless convection off the outside surface of a canister is cut off by deep sludge layers between canisters, a coarse in-canister rubble bed will not ignite at K Basins atmosphere temperatures. Rubble beds of fine particulate (< 0.1 mm) will ignite regardless of whether sludge is present or not. However, this is very unlikely since highly subdivided fuel will already be oxidized or surrounded by thick layers of oxide material.

This event is considered to be a beyond design basis accident.

6.2 SCENARIO DESCRIPTION

Starting Conditions and Assumptions--The central question regarding fuel heatup after dryout is whether, and to what extent, chemical reactions may occur between the fuel (or its corrosion products) and ambient basin air. This is because such reactions would be exothermic and result in fission product releases, whereas little fission product release would be expected in the absence of reactions. The original uranium metal fuel and one of its corrosion products, uranium hydride, are prone to such reactions under specific conditions.

A considerable number of studies and accidental occurrences were reported during the 1950's and the 1960's on the ignition of uranium under a

wide variety of conditions. In addition, the early work of Hartmann et al. (1951) showed that layers of uranium and uranium hydride powders can ignite spontaneously at room temperature. The occurrences of uranium ignition were often attributed, by hypothesis, to initiation by small amounts of surface hydride. This has led to the often made inference that fuel which could have corroded, when dried, will ignite in air.

Initiating Event and Sequence of Events--Postulated events that could be severe enough to significantly degrade the basin's capability to contain water were examined. Consequences of such events could potentially be extreme due to heat-up of fuel, loss of shielding, and boiling of remaining water with resulting airborne releases of radionuclides.

A postulated event might be the dropping of a major piece of equipment, such as a shipping cask or train, into the basin. However, Section 3.0 discusses the shipping cask drop and concludes that water loss from the basin could be replaced by make-up systems. The uncontrolled railroad train is an incredible event because of the design and procedural controls associated with basin operations.

As a result of a seismic event, failure of the basin is also postulated. The weak point of the basin is a construction joint in the concrete floor where the basins butt up against the adjacent reactor building foundations. An opening of this joint could result in basin leakage, a reduction in the basin water level, and a potential uncovering of the fuel and sludge.

Detection--As discussed in Sections 5.3.4 and 5.3.5 of the existing SAR, there are several diverse means available to detect reduced basin water inventory. Basin level instrumentation provides remote low level alarms which annunciate at both 105-K Basin facility control rooms and a remote location outside the facility. There are remote alarms for basin secondary cooling flow loss and local instrumentation to monitor secondary cooling pump suction and discharge pressure, which could indicate loss of pump suction from reduced level in the basin. There are several remote alarming gamma radiation monitors which could detect an increase in airborne activity.

6.3 FACILITY SPECIFIC BARRIERS

In the 105-K Basins, the construction joint in the bottom of the basin between the original reactor building structures and the basins have been determined by surveillance to lack rebar or doweling across the joints. The joints, filled with an elastic sealing compound, could spread apart during an earthquake and open a path to the soil for basin water.

A barrier, to prevent the escape of the basin water, is the use of seismically qualified steel-isolation barriers to seal off the area where the construction joint is located from the rest of the basin.

6.4 ACCIDENT PHENOMENOLOGY

Release--Ignition theory is the mathematical approach to address the potential for exothermic reactions for reasonable K-basin heatup conditions, and it will be applied in this section. Key inputs to the ignition theory are the decay power, heat loss mechanisms, and chemical reaction kinetics.

Decay power estimates were based on (Heard 1990).

The most important heat loss mechanisms, in terms of potential for limitation of fuel temperatures, are natural convection around the exterior and into the interior of canisters. From a purely thermal analysis of canister heatup (i.e., chemical reactions are not considered), it can be concluded that inclusion of natural convection along part of each fuel element results in a considerable reduction in the maximum fuel temperature versus a case with no such heat losses.

The maximum fuel temperature is important because of the exponential behavior of chemical reaction kinetics. It is important to appreciate that boundary conditions like the ambient temperature, relative humidity, and decay power have profound effects on the chemical reaction rates of uranium or its hydride. For this reason, experiences of fuel ignition shortly after discharge or after limited storage periods are not directly applicable to consequences of K-basin fuel dryout. Instead, these experiences must be put into context by application of reaction kinetic rate laws through ignition theory analyses incorporating specific situation conditions.

Factors influencing the oxidation rate of uranium are the lack of a protective oxide layer for the temperature range of interest, which determines the fundamental form of the reaction rate law, and the partial pressure of water vapor, with which the reaction rate increases.

Uranium ignition experience and theory is discussed in Epstein et. al. 1994. Ignition theory compares well with data for uranium foil ignition in pure air and with data for uranium powder ignition in oxygen. A comparison of the theory with early (1951) semi-quantitative data on the ignition of uranium hydride powder indicates that uranium hydride does not appear to be significantly more reactive than uranium. Uranium hydriding as a hypothetical cause of fuel element fires was suggested by these experiments and the fact that such fires started more readily in the presence of moisture. However, ignition theory together with a state-of-the-art uranium oxidation correlation, appears to be capable of rationalizing uranium fire experiences reported during storage of uranium metal turnings and briquettes without invoking the uranium hydride hypothesis.

Ignition theory is applied to K basin fuel elements in Epstein et. al. 1994 for the cases of intact and degraded fuel elements, respectively. The basic difference between these cases is that for intact elements the surface area for heat transfer is not too different than the surface area for chemical reaction, while for degraded elements in the form of a rubble bed, heat transfer occurs effectively on the exterior only and reactions may also occur on the interior, consequently ignition may be easier for rubble beds.

6.5 CONCLUSIONS

An approach for prediction of uranium metal and hydride ignition has been developed and validated, and it was applied to evaluation of the potential for fuel ignition following a postulated K basin dryout. This modeling approach represents a significant advance in the use of ignition theory for uranium metal and hydride, and for the first time supplies a consistent explanation for experiments carried out over several decades of research.

Classical ignition theory is applied to the boundary conditions specific to each situation using state-of-the art kinetic rate laws and property relations that were unavailable to earlier investigators. Good agreement is found with those experiments which are well characterized. The model is also consistent for anecdotally reported uranium metal fire experiences where the lack of specific quantitative information obviates the ability for rigorous comparison. It is further interesting to note that uranium fire experiences examined here do not require hydriding as the specific cause of ignition.

Natural convection heat transfer is the dominant heat removal mechanism for situations with relatively intact fuel and canister geometry. Experimental data and correlations are applicable to K basin geometry, supporting laminar natural convection external to the canisters (gases circulating from above the canisters in the basin down through the spaces between the canisters and back again) and laminar natural convection internal to the canisters (gases circulating from above open-top canisters down through the spaces between the fuel elements).

As a result of natural convection, ignition of fuel elements is not possible for closed-top canisters surrounded by less than about eight inches sludge. This conclusion applies to open-top canisters provided that similar quantities of non-fuel sludge do not occupy the canister interior. It is clear that actions to minimize the sludge between canisters in any configuration will reduce the potential for fuel ignition.

For the case of canister whose entire contents are fragmented debris, again as a result of natural convection, ignition is not possible unless the depth of sludge surrounding a canister is greater than about 10 percent of the debris bed height (which is about 50 cm deep). This holds true for internally-degraded canisters surrounded by similar internally-degraded canisters. When the neighbor canisters have intact fuel, the threshold sludge height for ignition is about 30 percent of the internal debris height; this is due to conduction into the K basin floor.

Ignition of degraded fuel in a canister not surrounded by significant sludge as in the case above, requires subdivision of the fuel into particle sizes smaller than about 1 mm. Such finely divided fuel is expected to have surface oxide from long-term corrosion in the K basin, and if this is true, then ignition would not be possible for fuel canisters containing fine debris.

The key conservatism embedded in the results is the assumption that all available fuel surface area is available for reaction. This assumption was made because the amount of exposed metal or hydride, as a result of degradation, the seismic initiator, or both, is presently unknown, and because layers of hydride have been hypothesized to exist which could suddenly become exposed. Cladding and oxide films probably would protect intact and degraded

fuel. Characterization program data which quantify the extent of such protection and the extent of hydriding are required for less conservative analyses.

A closed-top canister would not experience ignition and oxidation of its fuel content would only have limited consequences because the initially available reactant (oxygen and steam), when consumed, would cause only a trivial temperature rise. Further reaction would be possible only if the canister were failed, and typical puncture or tear failure areas would be small and would limit the supply of oxidant. Second, ignition of a thin surface layer of hydride or metal fines does not imply that remaining bulk or coarse fragments will react. A continued reaction requires sustained delivery of oxidant and sustained high (several hundreds of degrees C) temperatures.

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7.0 CRANE FAILURE AND CASK DROPPED ONTO FLOOR OF TRANSFER AREA

7.1 SUMMARY

A fully loaded transfer cask, dropped accidentally as the result of a crane failure, on to the floor of the transfer area from a height of 4.58 meters (15 feet) or more and overturning could result in the cask lid coming off and the irradiated fuel spilling out of the cask. An analysis of the radiological consequences from the release of volatile solids involving a fully loaded cask containing 3 canisters with 14 fuel elements in each canister of irradiated fuel was evaluated. In addition, the radiological consequences of the case where 3 canisters filled with sludge are dropped was also determined.

The potential dose to the onsite (Non-facility) worker was calculated to be 0.043 rem EDE and the offsite dose was calculated to be 0.015 rem EDE which makes this event the maximum credible accident. The overall risk associated with the cask drop can be evaluated by comparing the consequences of the potential accident to the risk acceptance guidelines given in WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*. For this event, the dose consequences fall within the radiological guidelines for an extremely unlikely event (annual probability less than E-04). The combination of probability and the dose consequences of this analysis are such that the risk acceptance guidelines are met.

7.2 SCENARIO DESCRIPTION

Starting Conditions and Assumptions--A cask containing three canisters of irradiated fuel, dropped onto the transfer area floor due to a crane failure, could empty its contents onto the floor with the potential for release of radioactive materials. A case where the cask containing three canisters of sludge from the basin floor was also analyzed.

The description of the cask dropped onto the floor of the transfer area accident begins with a listing of assumptions used for this section followed by the accident sequence.

Various assumptions are made about the system to model the accident scenario. These assumptions are as follows.

- It is assumed that the shipping cask contains its maximum of three canisters and each canister contains its maximum of 14 fuel assemblies for a total inventory of 42 fuel assemblies.
- Releases of radioactive materials can occur in the form of solids, liquids, or airborne releases. Since airborne contamination is the most prevalent health and safety risk, the accident analysis focuses primarily on airborne releases.

Development of the accident sequence was based on these assumptions.

Initiating Event and Sequence of Events--The accident scenario for the N-reactor fuel shipping cask assumes that a crane fails and the shipping cask drops from height of 4.58 m (15 ft) or more onto the hard surface of the

transfer area floor and opens exposing all of the fuel elements. It is assumed that if the impact opens the cask, then the three canisters of fuel inside the cask provide no protection for the fuel.

Detection--In the unlikely event that the cask were dropped from the crane, the operators would be immediately aware of the accident. In addition, such a cask drop could result in unshielding of the fuel and would cause a high radiation alarm requiring immediate evacuation of personnel.

7.3 FACILITY SPECIFIC BARRIERS

The 105-K Basins, are enclosed within a building that while not serving as a containment would confine or mitigate any releases occurring at the basins. While it is assumed in this scenario that if the cask falls, it will also open, this is very unlikely, since the cask lid has an internal locking mechanism operated by a detachable power tool and is not subject to activation by contact with items or protrusions on the floor or walls.

7.4 ACCIDENT PHENOMENOLOGY

Release From Damaged Fuel--An accident-release quantity determination was performed to establish the amount of contamination released from the facility (the released source term).

Information and assumptions used to determine the amount of material released are as follows:

- In the event of an accident where the cask is dropped from a crane onto a hard surface, it is assumed that the cask opens exposing all of the 42 fuel assemblies. Furthermore, it is assumed that each fuel element is broken once and exposes two fuel surfaces to the environment.
- For the modeling of a contaminated surface, it is assumed that there is a "depth of contamination" which when compared to the fuel element uranium mass and length will determine the fuel inventory at risk. Each broken exposed face of a fuel element is assumed to have a depth of contamination of 0.001 m from which radiological material can be released.
- Since the exposed fuel was modeled as a contaminated surface, the release fraction for a contaminated, non-combustible surface was used. This release fraction assumes that the fuel heats but does not melt or ignite (see section 5.4 and 6.4 of this report). The release fraction for respirable material made airborne due to a fire involving non-combustible materials is 1.0×10^{-5} . While Section 9.3.2.2.3 of the WRAP 1 PSAR provides the basis for this assumption, it should be recognized that some of the non-TRU isotopes with fairly low boiling points (especially Cesium) would have release fractions higher than $1 \text{ E-}05$. Actual known experiments determining release fractions for Cs and Sr are contained in BNWL-B-358, *Fractional Airborne Release of Strontium during the Combustion of 30 Percent Normal Tributyl in a Kerosine-Type Diluent* (Sutter, et al. 1974), and BNWL-B-274, *The Fractional*

Airborne Release of Dissolved Radioactive Materials During the Combustion of 30 Percent Normal Tributyl Phosphate in a Kerosine-Type Diluent (Mishima & Schwendiman 1973). In both experiments, the Cs and Sr were mixed with the Tributyl Phosphate (TBP) and kerosene. The maximum release fraction obtained was 2.5 E-03 for Cs and 2.2 E-03 for Sr. Therefore, as a bounding value for Cs and Sr, a value of 2.5 E-03 ARF should be conservative, considering that the burning scenarios would provide higher temperatures than the decay heat from the aged fuel.

- Quantities of fuel and radionuclides present were obtained from the Attachment.
- From Bergsman 1993, the longest N Reactor Mark IV fuel assembly was 0.66 m (26.1 inches) long.

The released source term is that quantity of contamination released from the K Basin during the accident sequence. The development of the source term that would be caused by a cask dropped onto the floor of the floor of the transfer area is as follows.

From Table 2-10 of the Attachment, the maximum fuel inventory in a Mark IV canister is 333.62 Kg. Therefore the material inventory for a cask containing 3 Mark IV canisters is:

$$3 \text{ canisters at } 333.62 \text{ Kg fuel/canister} = 1,000.86 \text{ Kg fuel}$$

The inventory at risk was determined by assuming that the exposed surface area of the fuel was the predominant factor involved in the release of the radioactive material. The estimate of the amount of fuel at risk assumed that each fuel element was broken in one place. For each fuel element this gives two exposed fuel surfaces each having a depth of Δt . The fraction of the fuel in an element that is L long with two exposed fuel surfaces each having a depth of Δt , that is at risk for release is:

$$\begin{aligned} \text{Fraction at Risk} &= \frac{2 \pi R^2 \Delta t}{\pi R^2 L} \\ &= \frac{2 \Delta t}{L} \end{aligned}$$

Thus, the fraction of the fuel in an element that is 26.1 in long with two exposed fuel surfaces each having a depth of 0.001 m or of the seven elements in a canister that is at risk for release is:

$$\begin{aligned} \text{Fraction at Risk} &= \frac{2 \text{ surfaces} \times 1.0 \times 10^{-3} \text{ m/surface}}{26.1 \text{ in} \times \frac{2.54 \times 10^{-2} \text{ m}}{\text{in}}} \\ &= 3.0169 \times 10^{-3} \end{aligned}$$

Then, modeling a contaminated solid surface, the fraction of fuel is multiplied by material inventory of a cask containing 3 Mark IV canisters. This yields the inventory at risk.

$$\begin{aligned}\text{Inventory at Risk} &= 3.017 \times 10^{-3} \times 1000.9 \text{ Kg} \times 10^3 \text{ g/Kg} \\ &= 3.02 \times 10^3 \text{ g fuel}\end{aligned}$$

The source term is calculated by multiplying the inventory at risk by the release fraction. Since the exposed fuel was modeled as a contaminated surface, the release fraction of 2.5×10^{-3} for a contaminated, non-combustible surface was used. This release fraction assumes that the fuel heats but does not melt. Thus, the released source term becomes:

$$\begin{aligned}\text{Released Source Term} &= 2.5 \times 10^{-3} \times 3.02 \times 10^3 \text{ g fuel} \\ &= 7.55 \text{ g fuel}\end{aligned}$$

Release Due to Aerodynamic Entrainment of Sludge--An accident release quantity determination was performed to establish the amount of contamination released from the facility (the released source term).

Information and assumptions used to determine the amount of material released are as follows:

- In the event of an accident where the cask is dropped from a crane onto a hard surface, it is assumed that the cask opens exposing all of the sludge.
- Quantities of radionuclides present in the sludge and sludge density were obtained from the Attachment.
- Based on recent measurements of the depth of sludge in K-East basin, the volume of sludge is estimated to be between 2047 to 2147 ft³ (57 to 61 m³). Since moist sludge should retain particles due to the large surface tension force, no entrainment of particles from wet sludge should occur for achievable wind speeds at the Hanford site (Epstein and Plys 1994). Only the dry sludge material near the sludge surface will be vulnerable for entrainment. Particles as large as about 1.0 mm may be suspended in the boundary layer that flows along the surface of the sludge. Owing to their weight, most of the larger particles are re-deposited into the sludge surface where they impact and free smaller surface particles. It is assumed that only sludge material within a depth of ten times the diameter of these larger particles will be at risk to entrainment. This is equivalent to a depth of about 1.0 cm of sludge and includes 12.7% of the total sludge volume.

The bounding release fraction for aerodynamic entrainment for a homogeneous bed of powder under static conditions within a structure is $4 \times 10^{-6} \text{ hr}^{-1}$ (Mishima 1993) and the release is assumed to occur over an 8 hour period.

The released source term is that quantity of contamination released from the K Basin during the accident sequence. The development of the source term that would be caused by a cask dropped onto the floor of the transfer area is as follows.

The volume of a canister which consists of two, open top, right circular cylinders, each with a diameter of 8 in and a height of 28 in is:

$$\text{Volume} = 2\pi R^2 H$$

$$= 2 \times 3.1459 \times (8 \text{ in})^2 \times 28 \text{ in} = 2,814.86 \text{ in}^3$$

$$\text{Volume} = 2,814.86 \text{ in}^3 \times 16.39 \text{ cm}^3/\text{in}^3 = 46,135.63 \text{ cm}^3$$

The material inventory for a cask containing 3 Mark IV canisters is:

$$3 \text{ canisters at } 4.61 \times 10^4 \text{ ml sludge/canister} = 1.38 \times 10^5 \text{ ml sludge}$$

The inventory at risk was determined by assuming that only the dry sludge material near the sludge surface will be vulnerable to entrainment. It was assumed that only sludge material within a depth of ten times the diameter of the larger particles will be at risk to entrainment. This is equivalent to a depth of about 1.0 cm of sludge and includes 12.7% of the total sludge volume.

Then, modeling a dried out surface of sludge, the total volume of sludge is multiplied by the fraction of the total that comprises the dry crust. This yields the inventory at risk.

$$\begin{aligned} \text{Inventory at Risk} &= 0.127 \times 1.38 \times 10^5 \text{ ml} \\ &= 1.75 \times 10^4 \text{ ml sludge} \end{aligned}$$

The source term is calculated by multiplying the inventory at risk by the release fraction. Since the sludge was modeled as a homogeneous bed of powder, the release fraction of $4.0 \times 10^{-6}/\text{hr}$ was used. The release is assumed to occur over an 8 hr period. Thus, the released source term becomes:

$$\begin{aligned} \text{Released Source Term} &= 4.0 \times 10^{-6} \times 8 \text{ hr} \times 1.75 \times 10^4 \text{ ml sludge} \\ &= 0.56 \text{ ml sludge} \end{aligned}$$

Consequences Due to Release--Using the released source term, the radiological doses to the onsite and offsite receptors were determined. The assumptions used to determine radiological consequences to human receptors are:

- The maximum offsite individual is located 12.0 km west of the release point. For the maximum offsite individual only short-term exposure effects were evaluated. For the short-term exposure effects case, the pathways that were analyzed included inhalation and submersion. Offsite ingestion is not included because in the case of an accident, any contaminated land or products would not be used. Therefore, ingestion would not be an actual exposure pathway. The maximum onsite individuals are located 4000 m east south east of the release point at the Hanford fire station (Refer to the Appendix). The maximum onsite exposure was assumed to be exposed via both submersion and inhalation. A 50-yr commitment period for the time the ingested material remains in the body was assumed.

- The ^{239}Pu is assumed to be in the oxide form for these calculations where the release is due to internal heating.
- From the attachment, the values of X/Q' for the selected receptor locations are presented in the table.
- The attachment recommends a value of 2.3×10^5 rem EDE/g for fuel. This recommended value incorporates both the EDE and limiting organ calculations.
- The attachment also recommends a value of 1.0×10^4 rem EDE/ml for as-settled sludge. This recommended value encompasses the range of variation in the sludge samples as well as incorporating both the EDE and limiting organ calculations.

The atmospheric dispersion coefficients and unit doses provided in the Attachment can be used to determine the effective dose equivalent resulting from a cask drop to the floor of the transfer area. This is done using the following equation from the Attachment.

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

where:

- Q = Quantity respirable radioactive material released (g or ml)
- X/Q' = Time integrated atmospheric dispersion coefficient
(Obtained from the Appendix Table 3-3)
- R = 3.3×10^{-4} m³/s standard man acute breathing rate (ICRP 1975)
- C = Dose per unit respirable radioactive material inhaled
(Obtained from the Attachment Table 2-16)

Consequences Due to Release From Fuel--Using this equation, the calculated EDE at the hanford boundary is 0.015 rem. Dose consequences calculated in the same manner for other selected receptors are given in the Table 7-1.

$$D = 7.6 \text{ g} \times 2.6 \times 10^{-5} \frac{\text{s}}{\text{m}^3} \times 3.3 \times 10^{-4} \frac{\text{m}^3}{\text{s}} \times 2.3 \times 10^5 \frac{\text{rem}}{\text{g}}$$

$$= 1.50 \times 10^{-2} \text{ rem EDE}$$

Consequences Due to Aerodynamic Entrainment of Sludge--Using the equation, the calculated EDE at the hanford boundary is 4.8×10^{-5} rem. Dose consequences calculated in the same manner for other selected receptors are given in the Table 7-1.

$$D = 0.56 \text{ ml} \times 2.6 \times 10^{-5} \frac{\text{s}}{\text{m}^3} \times 3.3 \times 10^{-4} \frac{\text{m}^3}{\text{s}} \times 1.0 \times 10^4 \frac{\text{rem}}{\text{ml}}$$

$$= 4.81 \times 10^{-5} \text{ rem EDE}$$

Table 7-1. - Summary of Calculated Dose Consequences

Receptor Location	X/Q^* (s/m ³)	EDE Fuel (rem)	EDE Sludge (rem)
100 m radius (100 m E)	1.2×10^{-2}	6.92	2.2×10^{-2}
Near River Bank (480 m NW)	5.6×10^{-4}	0.32	1.0×10^{-3}
Fire Station (4000 m ESE)	7.4×10^{-5}	0.043	1.4×10^{-4}
1994 Hanford Site Boundary (12,040 m W)	2.6×10^{-5}	0.015	4.8×10^{-5}

7.5 CONCLUSIONS

The potential dose to the onsite worker was calculated to 4.25×10^{-2} rem EDE. The offsite dose was calculated to be 1.50×10^{-2} rem EDE. The overall risk associated with the cask drop can be evaluated by comparing the consequences of the potential accident to the risk acceptance guidelines given in WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*. For this event, the dose consequences fall within the radiological guidelines for an extremely unlikely event (annual probability less than E-04). The combination of probability and the dose consequences of this analysis are such that the risk acceptance guidelines are met.

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8.0 SPENT ION EXCHANGE SHIPMENT FOR BURIAL

8.1 INTRODUCTION

One potential source of either respirable or ingestible radionuclides has been determined to be those isotopes that are dissolved in the water phase of an ion exchange column or module (IXC or IXM) (Marusich 1994). Dropping an ion exchange column or module could result in a column pressurization due to the piston action of the resin. Such a drop could break off a fitting or crack the column resulting in a spray release. This type of release can be used as a source term for the calculation of the onsite and offsite doses.

The radionuclide source term for this analysis is derived from reported analyses of K-Basin ion exchange column inventories by taking the highest reported values for ^{137}Cs , ^{90}Sr , ^{239}Pu , and ^{239}Pu for the ion exchange columns over the past five years. All the other Pu related radionuclides, ^{238}Pu , ^{241}Pu and ^{241}Am , can be obtained as ratios of the reported ^{239}Pu and ^{240}Pu values. The ^{90}Y source term was taken to be equal to the source term for ^{90}Sr .

An effective dose equivalent (EDE) was calculated for both the offsite and onsite receptors using the atmospheric dispersion coefficients and unit doses provided in the appendix and the following relationship:

$$D (\text{rem}) = Q (\text{Ci}) \times \frac{X}{Q'} (\text{s/m}^3) \times R (\text{m}^3/\text{s}) \times C (\text{rem/Ci})$$

where:

- Q = Curies respirable radioactive material released
- C = Dose per unit respirable radioactive material inhaled (rem/Ci)

The respirable quantity, Q, is derived from the inventory at risk and respirable release fraction.

After a spent column is taken off line, drained and stored, radiolysis breaks down the beads resulting in an accumulation of free water in the column and a reduction in efficiency of the resin of approximately 20 percent (Marusich 1994). Experience has shown that with age, spent resin in columns degrades resulting in the release of water i.e. free water shows up in the IX columns. The water does have a certain radionuclide concentration. It is also known that the resins show radiolytic degradation. The conclusion must be that the resin has reduced efficiency due to radiolytic degradation. Marusich reports that 20 percent of the column inventory can be accounted for in the released water.

The radionuclide inventory of a spray of water resulting from the drop of an ion exchange column is assumed to be 20 percent (0.20) of the inventory of the column.

Mishima reported spray releases from pressurized systems for pressures varying from 0.345 MPa (gauge) to 1.724 MPa (gauge) (50 psig to 250 psig) (Mishima 1993). For a pressure of 0.345 MPa (gauge) (50 psig) Mishima reports a respirable release fraction of 4 E-05. This analysis shows that the internal pressure generated by a drop of an ion exchange column from a height

of 3 meters (10 feet) is approximately 0.089 MPa (13 psi). Therefore, the release fraction for the lowest pressure reported in the Mishima study (0.345 MPa (gauge) [50 psig]) will be considered bounding (4 E-05).

8.2 CONCLUSIONS

The maximum onsite dose (Fire Station at 4000 m ESE) calculated was 3.1 E-04 rem effective dose equivalent (EDE) and 4.9 E-03 rem limiting committed organ (bone surface). The maximum dose to the offsite public was 7.7 E-05 rem (EDE) and 1.2 E-03 rem limiting committed organ (bone surface).

The calculated doses for this unlikely event are much less than the allowable 5 rem EDE and 50 rem limiting organ for onsite personnel and 0.5 rem EDE and 5 rem limiting organ for the offsite public. The allowable exposure limits are based on an annual frequency of occurrence of E-02 per year. The calculated frequency of occurrence is less than E-02 per year.

8.3 ANALYSIS

Mishima reports that for the depressurization of a volume containing a liquid phase at an initial pressure of 0.345 MPa, the respirable fraction of ejected water droplets is 4 E-05. It is assumed that the entire free water content of the resin is ejected during the depressurization process. Upon storage, radiolysis reduces the efficiency of the ion exchange resins. Marusich uses a retention efficiency of 80 percent of the on line efficiency in his analysis of the consequences of a hydrogen explosion in an ion exchange column (Marusich 1994). This means that 20 percent of the radionuclide inventory of a column that has been removed from service and drained, will be found in the water which accumulates in the column subsequent to disposal to provide the inventory at risk for the source term for the dropped ion exchange column accident analysis. This is a conservative estimate since a large fraction of the water inventory is bound in the resin beads and would not be ejected during the short duration impulse imparted to the resin as a result of dropping the column.

Based on 20% radionuclide inventory of an ion exchange column being at risk, the fraction of inventory in the respirable range released in with the spray release will then be:

$$\begin{aligned} R_f &= RF \times I_f \\ &= 4 \times 10^{-5} \times 0.20 \\ &= 8 \times 10^{-6} \end{aligned}$$

where RF is the release fraction reported by Mishima and I_f is the fraction of the ion exchange column inventory that is assumed to be in the column.

8.3.1 Internal pressure generated during a cask drop.

The internal pressure generated in an ion exchange column as a result of a cask drop is determined by calculating the specific impulse "J" imparted to

the column as a result of impact. The resin bed is assumed to be drained of excess water and to be in an aged state. Aging causes the resin to collapse due to radiolysis and gravitational factors resulting in free water in the column.

$$J = \int_{t_0}^{t_1} F dt = F \Delta t$$

or since $F = M \frac{dv}{dt}$ for a constant mass,

$$J = \int_{v_0}^{v_1} M dv = M \Delta v$$

where:

- F = average force developed from impact
- M = mass of resin bed
- $t_0 = 0$ = time at start of impact
- t_1 = time at end of impact
- v_0 = velocity at start of impact
- $v_1 = 0$ = velocity at end of impact

The pressure in the column due to the drop can be calculated by combining the previous expressions for the impulse. This gives the following equation.

$$\text{pressure} = \frac{F}{\text{area}} = \frac{M (v_1 - v_0)}{\text{area} (t_1 - t_0)}$$

Based on a 60% per weight water content from Table 8-1, the total weight of the resin can be calculated:

$$\begin{aligned} \text{weight} &= \left[45 \frac{\text{lb}}{\text{ft}^3} + \left(\frac{0.6}{0.4} \right) 45 \frac{\text{lb}}{\text{ft}^3} \right] * 5 \text{ft}^3 \\ &= 625 \text{ lb} \end{aligned}$$

Table 8-1. Ion Exchange Column Description

column length	5 ft
column diameter	17.5 in
cask height	6 ft
Low Boy truck height	3 ft
total resin slurry height (unaged)	5 ft
resin bulk density (dry)	45 lb/ft ³
resin volume (dry)	5 ft ³
water content of the drained resin bed (unaged)	60%

The first operation in the storage/disposal of spent ion exchange columns is the encasement of the spent column in a shipping cask. The cask is mounted in an upright position on a "low-Boy" eighteen wheel transporter. To determine the velocity of the column at impact, assume that the ion exchange column is raised high enough to clear the cask and then is dropped.

$$\begin{aligned}\text{drop height} &= \text{cask height} + \text{Low-Boy height} + \text{clearance} \\ &= 6 \text{ ft} + 3 \text{ ft} + 1 \text{ ft} = 10 \text{ ft}\end{aligned}$$

The time to impact and velocity at impact are found from the following equations:

$$h = v_o + \frac{1}{2} g t^2$$

$$v_t = v_o + at$$

With no initial velocity, the time to impact is 0.79 sec with an impact velocity of 7.8 m/sec (25.4 ft/sec).

Analytical data indicates that the ion exchange resins now being used at the K basin facility contain 60 percent by weight occluded water with a bed porosity of 40 volume percent (pg. 214, Brown 1950).

Assume that the cask drop collapses the resin to its void free state and that a fraction of the swelled beads also collapse. The height of the resin column will be reduced by 40-50 percent (2 - 2.5 feet [.6 - .75 m]) during the impact event. The resin column collapse of .6 m (2 ft) will be used in calculating the duration of the impulse. As the resin column collapses the average velocity of the center of mass decreases hyperbolically. Since the true velocity profile of the bed is dependent on the changing physical nature of the bed itself the bed velocity during impact is assumed to be the average of the velocities of the top and bottom of the bed at the time of impact.

$$\begin{aligned}
 V_m &= \frac{V_1 + V_0}{2} \\
 &= \frac{0 + 25.4}{2} \\
 &= 12.7 \frac{ft}{sec}
 \end{aligned}$$

The time duration of the impulse can be found from the following expression:

$$\begin{aligned}
 \text{impact duration} &= \frac{\text{distance}}{\text{velocity}} \\
 &= \frac{2 \text{ ft}}{12.7 \frac{ft}{sec}} \\
 &= 0.16 \text{ sec}
 \end{aligned}$$

Combining all the previous results, the force imposed on the bottom of the ion exchange column as a result of the drop can be calculated:

$$\begin{aligned}
 \text{pressure} &= \frac{F}{\text{area}} = \frac{M (v_1 - v_0)}{\text{area} (t_1 - t_0)} \\
 &= \frac{(625 \text{ lb}) (25.4 \frac{ft}{sec})}{(32.2 \frac{ft}{sec^2}) (1.67 \text{ ft}^2) (0.16 \text{ sec})} \\
 &= 1846 \frac{lb}{ft^2} = 12.8 \text{ psi}
 \end{aligned}$$

The respirable release fraction for the liquid ejected from the column is based on work done by Mishima (Mishima 1993). For conservatism a release fraction for a system pressure nearest to that calculated for the cask drop accident will be used. The lowest pressure studied by Mishima was 0.345 MPa (gauge) (50 psi). Mishima reports that the respirable release fraction for a release of liquid from a pressurized container at a pressure of 0.345 MPa (gauge) is 4×10^{-5} . The bounding respirable release fraction for a liquid ejected from a column subsequent to a drop is then 4×10^{-5} . The radiological release fraction is determined by taking the product of the respirable release fraction for the liquid and the fraction of the radionuclide inventory that is in the free water of the column which is 0.2 (Bauman 1966). The radionuclide release fraction is then.

$$(4 \times 10^{-5}) \times 0.2 = 8 \times 10^{-6}$$

8.3.2 Likelihood of Ion Exchange Column Drop Accident.

The information source used for the determination of the annual frequency for a column drop accident was IEEE Std-500 (ANSI 1984).

Assumptions and conditions:

- Human factors are not considered since Hanford cranes have fail safe devices that prevent "free wheeling" and bumping accidents are not considered to present a hazard.
- The probability of failure includes all modes of failure.
- The frequency of column drops will be based on moving the entire 60 column backlog inventory (stored in the 183 K-West chlorine vault).
- Only lifts of 10 ft or more will be considered.

The probability of a crane failure for all failure modes (i.e., clutch failures, cable breaks, tip overs etc.) is 1.0 E-04 per hour (ANSI 1984). It is estimated that the total time a column is suspended from a crane is approximately 45 minutes. Of the 45 minutes, approximately 40 minutes are required to paint the spent column prior to storage or final disposition. During the painting process the column is suspended approximately 6 ft above the ground, a drop of 1.8 m (6 ft) is not considered to be sufficient to damage a column. Based on operating experience, it is estimated that 5 minutes are required to lift a column after painting and lower it into a transportation cask. For conservatism, twice the average lifting time will be used, lift time = $2 * 5 \text{ min} = 10 \text{ min}$.

A total of 60 columns could potentially be transported during the first year, therefore there is a total 600 minutes or about 10 hours per year lift time to a height of 3 meters (10 feet) or greater.

The annual frequency for a column drop is then

$$1 \times 10^{-4} / \text{hr} \times 10 \text{ hr/yr} = 1 \times 10^{-3} / \text{yr}.$$

Referring to WHC-CM-4-46 (section 7.0, Rev. 3, page 4, Table 1), an annual frequency of 10^{-3} falls in the unlikely category.

8.3.3 Dose Consequences For Onsite and Offsite Receptors.

For the purpose of dose consequence calculations, the maximum ^{137}Cs , ^{90}Sr and $(^{238} \text{ thru } ^{241})\text{Pu}$ plus ^{241}Am loadings for ion exchange columns were obtained from the inventory data sheets (Marusich 1994). It is assumed that a single hypothetical column exists that contains the maximum number of curies for each of the inventoried isotopes. As such, the hypothetical column would contain 0.9 curies of $^{239/240}\text{Pu}$, 87.8 curies of ^{137}Cs , and 94.1 curies of ^{90}Sr .

Table 8-2 summarizes the unit curie dose consequences (rem/Ci) for the radionuclides report for the ion exchange column (see summary Table 2-11 from the attachment). The unit dose for the $^{239/240}\text{Pu}$ is derived from the total K West basin fuel total inventory in the following manner:

$$\begin{aligned} \text{EDE(rem/Ci)} &= \frac{\text{EDE(rem/gram)} (\text{total weight of fuel})}{\text{total curies of Pu}^{239/240}} \\ &= \frac{(1.4 \times 10^5 \frac{\text{rem}}{\text{g}}) (952,198 \text{ kg})}{(1.04 \times 10^5 \text{ Ci Pu}^{239} + 5.4 \times 10^4 \text{ Ci Pu}^{240})} \\ &= 8.4 \times 10^8 \text{ rem/Ci} \end{aligned}$$

On a per curie basis, the contribution to the radiological dose from either strontium or cesium is three orders of magnitude less than the plutonium contribution. Therefore, only the plutonium is used in the calculation. Calculations for both the EDE and the limiting committed organ dose equivalents require a breathing rate ($3.3 \times 10^{-4} \text{ m}^3/\text{sec}$) and atmospheric dispersion constants (see Table 8-3 for χ/Q').

For example, the calculated EDE for a point source release for the onsite receptor (Fire Station, 4000 m ESE) is;

$$\begin{aligned} D (\text{rem}) &= Q (\text{Ci}) \times \text{RF} \times \frac{\chi}{Q'} (\text{s/m}^3) \times R (\text{m}^3/\text{s}) \times C (\text{rem/Ci}) \\ &= (0.9 \text{ Ci}) (8 \times 10^{-6}) (1.47 \times 10^{-4} \frac{\text{s}}{\text{m}^3}) (3.3 \times 10^{-4} \frac{\text{m}^3}{\text{s}}) (8.4 \times 10^8 \frac{\text{rem}}{\text{Ci}}) \\ &= 2.9 \times 10^{-4} \text{ rem} \end{aligned}$$

and the limiting organ dose for the same receptor is;

$$\begin{aligned} D (\text{rem}) &= Q (\text{Ci}) \times \text{RF} \times \frac{\chi}{Q'} (\text{s/m}^3) \times R (\text{m}^3/\text{s}) \times C (\text{rem/Ci}) \\ &= (0.9 \text{ Ci}) (8 \times 10^{-6}) (1.47 \times 10^{-4} \frac{\text{s}}{\text{m}^3}) (3.3 \times 10^{-4} \frac{\text{m}^3}{\text{s}}) (1.2 \times 10^{10} \frac{\text{rem}}{\text{Ci}}) \\ &= 4.2 \times 10^{-3} \text{ rem} \end{aligned}$$

Table 8-3 lists the EDE and limiting committed organ doses for the several identified receptor locations.

Table 8-2. Unit Consequences for Selected Isotopes

Isotopes	Effective Dose Equivalent (rem/Ci)	Limiting Organ (rem/Ci)
$^{90}\text{Sr}/^{90}\text{Y}$	2.7×10^5	
$^{137}\text{Cs}/^{137\text{m}}\text{Ba}$	3.2×10^4	
$^{239/240}\text{Pu}$	8.4×10^8	1.2×10^{10}

Table 8-3. Calculated Dose Consequences

Receptor Location	χ/Q' (s/m ³)	EDE (rem)	Limiting organ (rem)
100m radius (100 m E)	7.3×10^{-2}	1.5×10^{-1}	2.09
Near River Bank (480m NW)	2.2×10^{-3}	4.3×10^{-3}	6.1×10^{-2}
Fire Station (4000m ESE)	1.5×10^{-4}	2.9×10^{-4}	4.2×10^{-3}
1994 Hanford Site Boundary (12,040m W)	3.6×10^{-5}	7.2×10^{-5}	1.20×10^{-3}

9.0 HYDROGEN DEFLAGRATION IN ION EXCHANGE MODULES AND FILTERS

BACKGROUND - The operation of KE and KW Basins for interim storage of irradiated N reactor fuel requires the use of mixed bed, Duolite ARN-9381®, ion exchange modules, ion exchange columns, and filter cartridges to remove radionuclides and particulates from the basin water. The ion exchange modules are used for normal concentrations and the ion exchange columns are used only if the concentration is large enough that the modules cannot handle the load. The cartridge filters are run until the dP across the filter exceeds a predetermined point and the filter is exchanged.

Loaded ion exchange modules, ion exchange columns, and plugged filters are removed from service, drained for 24 hours, and placed in storage. Ion exchange columns are stored in lead caves, spent cartridge filters are stored outside 105 K in concrete filled culverts. When a few columns are in storage a railroad car containing a burial box is brought in and they are removed from the storage cave and placed in the burial box. The cover is then placed on the burial box and the railcar is moved to the 183-K Building vault until it is needed again. 183-K Building currently contains two burial boxes, one box contains 33 used columns and the other contains six used columns.

The radionuclides in the spent resin result in the generation of hydrogen gas in the resin bed as a result of radiolysis of the water and the resin beads. Radiolysis of entrapped water also occurs in the cartridge filters. The hydrogen diffuses out of the bed and throughout the open spaces in the column and cartridges and could pose a threat in the event of a deflagration or detonation of a significant accumulation.

ANALYSIS - The potential for a hydrogen deflagration/detonation in an ion exchange column was studied (Marusich 1994) and consequences determined for accidents involving ion exchange modules, ion exchange columns, and cartridge filters.

An ion exchange column is 150 cm in height, 45.7 cm Sch 10 pipe with 0.64 cm walls and 44.5 cm inner diameter. The bed occupies the middle 104 cm when the beads have absorbed water and swelled. It is designed for an operating pressure of 0.52 MPa (gauge) and tested at 0.86 MPa (gauge). The burst strength of 45.7 cm Sch. 10 pipe is about 20 times greater based on ultimate strength and hoop stress.

Six ion columns are cast in a concrete block to form an ion exchange module. The ion exchange modules are made of 40.6 cm, Sch. 30 pipe, 71 cm long. With the piping at each end the total column height is 106.7 cm. The internal diameter is 38.74 cm.

The cartridge filters are 76.2 cm in diameter and 83.8 cm long containing a polypropylene fiber wound on a 304 stainless steel core which is resistant to radiation and acids.

During operation the ion exchange columns remove radionuclides from the basin water. The primary isotopes removed from the water are ^{90}Sr , ^{137}Cs , and transuranics, TRU. Marusich shows using inventory estimates from the "worst" ion exchange column, ID numbers 203-205 containing 94.1 Ci ^{90}Sr , 0.6 Ci ^{137}Cs , and 0.35 Ci TRU, that the hydrogen generation rate is approximately 0.003

1/H₂/hr. Over an extended period of time, several years, this hydrogen could accumulate in the columns and in the burial box.

The potential for H₂ deflagration/detonation was evaluated for the three different columns and filters using various assumptions for their position in the burial box and potential for concentration of H₂ within the columns and burial box.

CONSEQUENCES - Marusich 1994 determined that the worst case consequences resulted from an estimated 30% hydrogen concentration in an ion exchange column at the end of 3.9 years. A spark was assumed to occur as a result of sudden column movement and ignites the hydrogen in the open space within the column. The burn pressure for 30% hydrogen is 117 psi and the peak reflected pressures due to reflection of the shock could reach 28 atmospheres or 2.75 MPa. Since the gas detonates it is assumed that a seam on a weld opens. The resin and water expelled are assumed to behave like a liquid. Only 20% of the radionuclides are removed from the resin due to radiolysis. These are assumed to be contained within the liquid and not reabsorbed elsewhere within the resin bed. The release is assumed to resemble a spray and using Section 3.2.3 of Mishima 1993 the release fraction value of 2×10^{-4} is used to bound the scenario. The release is therefore:

$$(0.2)(2 \times 10^{-4}) = 4 \times 10^{-5} \text{ of the contents of the column}$$

Unit doses were calculated in Marusich 1994 and are shown in Table 9-1.

Table 9-1. Unit Doses for Hydrogen Deflagration at K Basins.

<u>Radionuclide</u>	<u>Onsite, rem</u>		<u>Offsite, rem</u>	
	<u>EDE</u>	<u>Organ</u>	<u>EDE</u>	<u>Organ</u>
⁹⁰ Sr	3.6 E-02	4.0 E-01	2.9 E-03	3.2 E-02
⁹⁰ Y	1.2 E-03	4.9 E-03	1.0 E-04	4.0 E-04
¹³⁷ Cs	4.9 E-03	4.9 E-03	1.0 E-04	4.0 E-04
TRU	65.9	1,154.3	5.3 E+0	92.3 E+0

For the dose consequences Table 9-2, it is assumed that 4 E-05 of the column quantity is released. Worse case columns, 203-205 contain:

94.1 Ci ⁹⁰Sr
0.6 Ci ¹³⁷Cs
0.35 Ci TRU

Column 225 contains:

13.0 Ci ⁹⁰Sr
32.0 Ci ¹³⁷Cs
0.66 Ci TRU

Table 9-2. Dose Consequences, EDE, rem, for Hydrogen Deflagration.

Receptor Location	$X/Q's$ (s/m^3)	IXC (rem)	IXM (rem)	EDE Filters (rem)
100 m radius (100 m E)	7.3×10^{-2}	2.9×10^{-1}	4.5×10^{-1}	4.5×10^{-1}
Near River Bank (480 m NW)	2.2×10^{-3}	8.8×10^{-3}	1.4×10^{-2}	1.4×10^{-2}
Fire Station (4000 m ESE)	1.5×10^{-4}	6.0×10^{-4}	9.2×10^{-4}	9.2×10^{-4}
1994 Hanford Site Boundary (12,040 m W)	3.6×10^{-5}	1.4×10^{-4}	2.2×10^{-4}	2.2×10^{-4}

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10.0 RELEASE OF CHLORINE

10.1 BACKGROUND

Chlorine, used for water disinfection of the K Basin water, is normally injected into the water supply obtained from the river. The PHA identified six potential events involving the release of chlorine from the 183-KE water treatment facility. The first three events, grouped together because of their similarity, involved a release of a chlorine water mixture backflowing from the water treatment system at the Columbia River intake. The second set of three, also combined as one accident scenario, involved the rupturing of a single chlorine cylinder. The overriding concern is a death or injury as a result of inhaling chlorine fumes. A complete analysis of chlorine releases is presented in Morris 1991. Since the analysis was performed, the single 908 kg (2000 pound) chlorine cylinder system has been replaced by one using four 68 kg (150 pound) cylinders. The existing analysis is still bounding, because it shows that the cylinder failure affects a significantly larger area than the chlorine backflow area and the nearly instantaneous release of 908 kg (2000 pounds) would exceed that of the combined four cylinders of 68 kg (150 pounds) each.

10.2 ANALYSIS

This analysis estimates the downwind, centerline, groundlevel concentration of chlorine vapor in the air resulting from two proposed accident scenarios: an instantaneous rupture of a full cylinder, containing 908 kg (2000 lbs) of liquid chlorine, and the accidental, continuous backflow of chlorine from the water treatment facility to the river, at a rate of approximately 5.7 kg/hr (12.5 lbs/hr). The primary areas of concern included the potential exposure of plant personnel to dangerously high chlorine levels, and the possible public exposure to the vapors via the relatively close Columbia river which is approximately 915 meters (3000 feet) away.

Chlorine, a vapor at normal atmospheric conditions (21 °C [70 °F] and 14.7 psia) has a normal boiling point of -29.3 °F. Storage of chlorine in the cylinder at atmospheric temperature (21 °C [70 °F]) requires a pressure of approximately 0.827 MPa (gauge) (120 psig). When this material is released to the atmosphere, it will quickly flash to a vapor, drawing heat from the surroundings, i.e. the air, ground, tank contents, etc.. This evaporation will result in the formation of water condensation in the atmosphere, and the cool gases will fall to the ground. In addition, due to chlorine's higher molecular weight (70.9), as compared to air (28.8), the denser vapors will also tend to slump toward the ground. This vapor will then be dispersed by wind and air movement, gravity and diffusion.

The instantaneous failure of a cylinder would result in a two phase mixture of vapor and liquid being pushed out by the internal pressure, 0.827 MPa (gauge) (120 psig). A cylinder failure involving this pressure might cause vapor jetting, cylinder projectiles and liquid droplets being thrown out, resulting in increased vapor dispersion. As a simplifying assumption, these effects were ignored, and the release was treated as an instantaneous volume release of a liquid/vapor mixture. The volume of this mixture was estimated based on the mixed density, and an assumed 2 meter release elevation.

Chlorine, used for water disinfection, is normally injected into the water supply from the river at a rate of (5.7 kg/hr, [136 kg/day] (12.5 lbs./hr, [300 lbs./day])). A combination of failures, including an operator error to not shut off the chlorine supply and failure of valves/pumps to prevent backflow to the river, is necessary to produce this scenario. It was assumed that the duration of this error could occur for 8 hours (1 shift). It was also assumed that the chlorine that flowed to the river would completely and immediately evaporate into the atmosphere, in order to maximize the air release. It is more likely that the chlorine would be diluted by the river water, at the submerged intake, and would pose a greater environmental threat to the local aquatic life than to the public.

These scenarios were used to estimate the potential ground level, downwind, chlorine concentration using the SLAB dense gas dispersion code. Dense gases or vapors are material whose density is greater than air. These materials may rise slightly upon release, according to their momentum. However, their densities cause them to fall to the ground, following the land contours and dispersing slowly. The SLAB code accounts for the physics of heavy gas dispersion as well as normal atmospheric and turbulent dispersion processes.

The SLAB code has several simplifications included in its calculational routines. One limitation is the inability to account for variations in land elevations and buildings; these features can affect actual dispersion by providing low points or air pockets of limited mixing where the dispersion is reduced and concentrations could be greater than expected. On the other hand, increased surface roughness, in the form of structures, shrubs, etc. can produce increased turbulence and mixing. This analysis has assumed a surface roughness similar to level desert.

Perhaps the most significant assumptions in this analysis are selection of meteorological conditions throughout the dispersion. The constant meteorological dispersion assumption is a limitation of the SLAB code, and other similar codes. Due to the lack of actual site meteorological data, the analyses were performed for each release condition at a single meteorological condition, consistent with NRC emergency planning practices.

Finally, the calculated concentrations are based on a one minute averaging time for the cloud located at a given distance from the release point. The choice of a one minute average was due to the short duration of the instantaneous release, for which longer averages would incorporate time periods which did not include the transient chlorine cloud. Care should be taken in comparing these results to published standards such as: the Emergency Response Planning Guideline, ERPG, (one hour averages) and the Immediate Danger to Life and Health concentration, IDLH, (30 minute averages). These values for chlorine are: ERPG-2 value of 3 ppm, exposures below which nearly all individuals would come to no permanent harm after an one hour exposure, ERPG-3 value of 20 ppm, exposures below which individuals exposed for up to one hour are not expected to receive health threatening effects, and IDLH value of 30 ppm, for which exposures of less than thirty minutes are not felt to be harmful.

As the cloud moves down wind, it spreads and the duration at any given location increases. Therefore, when the duration exceeds one hour (or 30 minutes for the IDLH level), use of ERPG's may be considered, based on the actual exposure duration and possible evacuation of exposed personnel. For

these releases, the SLAB code determines the time, after release, when the peak concentration occurs at a downwind location, the peak average concentration, centered on that time. Therefore the average calculated value represents the maximum average and the cloud will not have that maximum value for the entire duration of time it passes over the location of concern. The code also determines the total time duration during which the cloud will pass over the location.

Table 9-1 presents a summary of the dispersion analyses performed, including the distance from the release to the ERPG levels, ERPG-2 and ERPG-3, and the IDLH level, the time it takes for the peak concentration to occur at the given ERPG distance, and the duration of the cloud at that location.

The analyses results indicate that cylinder failure has a significantly larger impact area than chlorine backflow. The continuous release can produce locally higher concentrations (48900 ppm) at the river, assuming complete evaporation, but the chlorine dissipates rapidly resulting in river concentrations on the same order of magnitude as the cylinder rupture scenario within 15 meters of the intake location and concentration below 3 ppm within 750 meters.

The dispersion analysis results indicate that the instantaneous release, due to the large amount of chlorine could produce the most serious, i.e. greatest potential area of high chlorine concentrations. The concentrations at the river were estimated at approximately 1100 ppm with a duration of 17 minutes, compared to an IDLH value (30 minute reference) of 30 ppm. The continuous release can produce locally higher concentration (48900 ppm) at the river, i.e. the water intake location, assuming complete evaporation, but the chlorine dissipates rapidly resulting in river concentration on the same order of magnitude as the cylinder rupture scenario within 15 meters of the intake and concentrations below 3 ppm within 750 meters. These results indicate that these accident scenarios do present possible exposure concerns to both onsite personnel and the public. Evaluation of the risks associated with these potential release scenarios should include both consideration of the consequence levels, i.e. associated with public/personnel exposure, and the probability of the combination of conditions necessary to produce that consequence level.

10.3 SUMMARY AND CONCLUSIONS

The Immediately Dangerous to Life and Health (IDLH) value for chlorine is 30 ppm. Both of the chlorine release scenarios yield chlorine concentrations at the site boundary that are greater than the IDLH value. However, the dispersion analyses indicate that the instantaneous release of chlorine has a much greater impact than the chlorine backflow scenario. Site workers are at greater risk due to their proximity to the point of release and due to the fact that the cloud can reach them within a few minutes.

The complete cylinder failure results in significant concentrations of chlorine at the river in about ten minutes. Levels of 30 ppm (IDLH value), for which exposures of less than thirty minutes are not felt to be harmful, extend almost 4.8 kilometers (3 miles) from the release point and peak at this value in 1.1 hours of the event. Levels of 20 ppm (ERPG-3), for exposures below which individuals exposed for up to one hour are not expected to receive health threatening effects, extend about 5.6 kilometers (3.5 miles) and peak

at this value some 1.4 hours after the event. Levels of 3 ppm (ERPG-2), for exposures below which nearly all individuals would come to not permanent harm after a one hour exposure, extend to slightly more than 14.5 kilometers (9 miles) and peak at this level some 3.1 hours after the event.

The use of 0.907 metric ton (one ton) chlorine cylinder at the Hanford Site water treatment facilities has been recently reviewed (Schade 1991). This review concluded that the risks associated with 0.907 metric ton (one ton) chlorine cylinders can be maintained at an acceptably low level by the conscientious application of Chlorine industry standards for facility design and construction, product handling, operations, and emergency preparedness measures (Schmidt 1991).

The basic requirements for protection of the public from a potential chlorine release at the 183-KE water treatment plant are in place. Implementation of the recommendations contained in Schmidt 1991 are appropriate to help improve and assure the timeliness and effectiveness of emergency response by both onsite personnel and the public.

It is important to consider a number of factors in determining the extent of actions required for protection against chlorine hazards. Among them are that chlorine is among the top ten chemicals produced and utilized by the industrial world. While fatalities have occurred as a result of incidents involving exposure to chlorine, most involve relatively small releases in confined areas. In the twentieth century, only one of the fatalities from a chlorine event has occurred farther than 450 yards from the event site (Morris 1991).

Based on the widespread use of chlorine in the public domain, and consistent with the provisions of DOE Order 5480.1B, the subject application of chlorine is judged to be a hazard routinely accepted by the vast majority of the public. The risks associated with the use of this product for water treatment on the Hanford Site can be maintained at an acceptably low level by the conscientious application of chlorine industry standards for facility design and construction, product handling, operations, and emergency preparedness measures. Structural design would be in accordance with the Uniform Building Code and the design loadings as defined in SDC-4.1(DOE-RL 1989). Based on the classification as a commonly accepted hazard, and the requirement for the application of existing industry standards, additional safety analysis documentation is not necessary.

Table 10-1. Summary of SLAB Analysis Cases Ground Level,
Downwind Centerline Concentrations

Cylinder Rupture

	F Stability 2.2 mph
Distance to 30 ppm, (IDLH)	4,392 meters (14400 feet)
Time of Peak Concentration	1.1 Hr.
Duration of Cloud	1750 Sec.
Distance to 20 ppm, (ERPG-3)	5,434 meters (17815 feet)
Time of Peak Concentration	1.4 Hr.
Duration of Cloud	1930 Sec.
Distance to 3 ppm, (ERPG-2)	14,894 meters (48834 feet)
Time of Peak Concentration	3.1 Hr.
Duration of Cloud	6420 Sec.
Concentration at River	1102 ppm

Chlorine Backflow to Columbia River

	F Stability 3.5 kmph (2.2 mph)
Distance to 30 ppm, (IDLH)	168 meters (550 feet)
Time of Peak Concentration	Note 1
Duration of Cloud	8 Hr.
Distance to 20 ppm, (ERPG-3)	218 meters (715 feet)
Time of Peak Concentration	Note 1
Duration of Cloud	8 Hr.
Distance to 3 ppm, (ERPG-2)	704 meters (2310 feet)
Time of Peak Concentration	Note 1
Duration of Cloud	8 Hr.
Concentration at River	48900 ppm

Note (1) - A steady state release with an eight hour duration was assumed, therefore these concentrations would exist for the entire period.

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11.0 POWER AVAILABILITY AND RELIABILITY

Each irradiated fuel pool cooling and cleanup system is a manually controlled system that can be shut down for periods of time for maintenance or replacement of malfunctioning components. The system has some redundant equipment (e.g., two cartridge filters, two primary pumps, two separate ion exchange systems, etc.). All electrical power supply is from a single source, however.

During any power outage all normal fuel handling activity will cease, since normal instrumentation will be lost. Radiation levels, coolant temperature, and coolant level can be determined with portable instruments or other means. Monitoring is not necessary for temperature control because temperature will never reach boiling as described below.

A series of analyses (Barrington 1994) were performed to determine if the fuel storage basins could exceed 100 °F for a series of postulated operational scenarios.

To obtain accurate estimates of basin temperatures, lumped heat capacity, multi-node, thermometric models of the 105 KE and KW facilities were constructed. The resulting models were verified against historical data and shown to replicate the observed basin temperatures with a standard error of less than 1 °F.

Various "worst-case" operating scenarios were investigated using the models. The worst-case scenario is defined as operation with the recirculation and skimmer pumps active, without exhaust fans, and without evaporation.

Maximum pool temperatures of 156.4 and 186.3 °F were predicted for 105 KE and KW, respectively, without evaporation. The corresponding values with evaporation were 97.4 and 85.3 °F, respectively.

A lack of cooling would arise if the power outage occurred during the short time interval when the crane lifts the cask from the well car and lowers it into the basin. However, based on current heat loads (7 to 23 year old fuel) and using the methods of (Buchanan 1980) the water temperature will never reach boiling therefore no emergency cooling or power is required.

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12.0 ASHFALL

The Hanford reservation is in a region subject to ashfall from volcanic eruptions. The three major volcanic peaks closest to the project are: Mt. Adams about 161 kilometer (100 miles) away, Mt. Rainier at about 177 kilometers (110 miles) away, and Mt. St. Helens approximately 209 kilometers (130 miles) away.

Important historical ashfalls affecting this location were from eruptions of Glacier Peak about 10,000 BC, Mt. Mazama about 4000 BC, and Mt. St. Helens about 6000 BC. The most recent ashfall resulted from the May 18, 1980 eruption of Mt. St. Helens. As a result of the 1980 Mt. St. Helens eruption, the site design criteria were modified to include ashfall.

Although the K basin roofs are classified as Safety Class 3 (Langevin 1991), the structure has to meet certain Safety Class 1 requirements because of three-over-one (3/1) concerns. If a collapse of the superstructure were to occur, significant damage to the Safety Class 1 basins and fuel canisters could occur (Kanjilal 1994). Kanjilal further concluded that the three-over-one concerns would exist if certain recommended fixes to strengthen the superstructure were made. These fixes have been incorporated.

SDC-4.1, Revision 11 pertinent sections state:

- These criteria establish the design loads and acceptance criteria for all facilities and new additions or for modifications of existing facilities at Hanford. Deviations from the natural phenomena design loads shall be approved by DOE prior to implementation. All deviations shall be justified in the design report.
- Safety Class 1 structures, systems and components, (DOE Order 6430.1A Safety Class), are those that perform a function required for nuclear criticality safety or whose failure might result in a significant release of radioactive, hazardous, or toxic materials as defined by DOE Order 6430.1A, Division 13, as it relates to non-reactor nuclear facilities. Included in this category are those systems and components vital for safe shutdown. Safety Class 1 structures shall be designed for normal loads, as well as wind loads, and roof loads. Load combinations and allowable stresses for these loads shall be as noted in Section AH. Additionally, these Class 1 structures must meet the following natural phenomena loading. The loadings are differentiated between reactor and non-reactor facilities.

Non-Reactor Safety Class 1 structures, systems and components are to withstand the effects cited above for reactor facilities with the exceptions: 1) SSE is identified as the Design Basis Earthquake (DBE); 2) OBE is not required; 3) Design Basis Tornado is replaced by Design Basis Wind, and 4) Other abnormal loads considered on a case by case basis.

Design Basis Earthquake (DBE) for Non-Reactor Structures. Response spectra for the design ground motion for the Design Basis Earthquake Non-reactor Safety Class 1 Structures are specified. These design response spectra are applied for the horizontal directions in the free field. For

vertical motion, the design response spectra shall be taken as 2/3 of the horizontal spectra over the entire range of frequencies. Existing structures may be evaluated using methods defined in UCRL-15910 (Kennedy et. al. 1990), Section 4.4.1. An Operational Basis Earthquake (OBE) is not required for non-reactor structures.

Design Basis Ashfall. A volcanic ashfall is considered to be a potentially occurring natural phenomenon in the category of tornado and earthquake. All new Safety Class 1 structures, systems and components are required to meet the design basis ashfall criteria as follows:

Thickness of the ashfall:

11.4 cm (4.5 inches) uncompacted
7.6 cm (3 inches) compacted

Duration and rate of ashfall:

Duration of ashfall	20 hours
Maximum ashfall rate	1.14 cm (0.45 inches) per hour for 6 hours
Average ashfall rate	.38 cm (0.15 inches) per hour

Design ashfall loading:

24 psf dry, compacted

Density of ash:

Dry, loose	72 pcf
Dry, compacted	96 pcf
wet, compacted	101 pcf

Ashfall loading is to be combined with snow load. Ashfall in combination with snow loading will normally produce the highest loading for design. Ashfall loading is not to be combined with earthquake.

Roof Loads. Snow loads shall be in accordance with ANSI A58.1 (ANSI 1982), Section 7. Seventy three kilograms per square meter (kgs/m^2) [fifteen pounds per square foot (psf)] shall be used as the ground snow load, PG, but in no case shall roofs be designed for less than 97.6 kgs/m^2 (20 psf) snow load.

Internal memo 92-RAG-029 (Giller 1992), dated August 14, 1992, is a calculation showing the allowable roof load for different panel span lengths. The calculation has been peer reviewed and signed. The reported allowable panel load is based on breaking strength information from the original 1952 construction specification with appropriate factors of safety applied.

WHC-SD-N031-SA-002 (Kanjilal 1994), Rev 0, states "Report WHC-SD-NR-SA-024 (Winkel 1994), 2/18/94, was issued to document the evaluation of seismic adequacy of the water retention boundaries of the 105-KE/105-KW reinforced concrete fuel storage basins for an additional 20 years of service. As U. S. Department of Energy (DOE) Safety Class 1 facilities (K Area Systems- Identification of System Numbers and Safety Classifications, WHC-SD-NR-SD-003

[Wilson 1994), Rev. 0), seismic adequacy of the storage basin structures is defined in DOE Order 6430.1A and UCRL-15910. If adequacy of the existing structures could not be demonstrated, modifications were to be identified that would bring the facilities into compliance with the appropriate requirements."

WHC-SD-N031-SA-002 (Kanjilal 1994) further states "An additional objective of the evaluation was to evaluate the seismic adequacy of the steel-framed superstructures immediately above the storage basins. The steel frame superstructures are classified as Safety Class 3 facilities (WHC-SD-NR-SD-003 [Wilson 1994]), but an evaluation for the Safety Class 1 Design Basis Earthquake (DBE) loading is necessary because of the three-over-one (3/1) concern. That is, if a superstructure collapse should result from a Safety Class 1 DBE, significant damage to the Safety Class 1 basins and fuel canisters could occur."

Carrying this to its logical conclusion, the roof also has to show structural adequacy for the snow plus ashfall roof load requirement. The seismic requirements for the non-reactor facilities definitely do not require application of ashfall plus snow plus seismic forces.

As a result of the 1980 Mt. St. Helens eruption, the site design criteria were modified to include ashfall. The 105-KW roof was replaced in 1988 due to concern for degradation of original roof panels at that facility. The 105-KW roof has been analyzed and will survive the limiting live load combination consisting of the design basis snow and ashfall loads of the current site design criteria (Frier 1993b).

Deterioration of the 105-KE roof structure membrane has occurred and is to be replaced as part of the N Reactor Facilities Roof Repair Program Management Plan (Frier 1993a). The replacement will not change the loading criteria, so the original Uniform Building Code Criteria involving live snow and seismic loads remain applicable. Currently, the roof will not meet SDC-4.1 Rev 11 requirements for Non-Reactor Class 1 structures for combined snow and ash loading (Giller 1992) without reducing the roof panel effective span length to .99 meters (3.25 feet) with additional support structure. This is not a significant safety concern, as the basic superstructure will support the loads and the failure of roof panels would only result in entry of snow and ash into the basin area. There is no significant effect from snow and ash entry on the storage configuration of the fuel relating to criticality concerns. Although the presence of ash in the basin water may challenge the filtration system, as has been demonstrated previously, even loss of basin recirculation is not a short term problem. Actions can be taken to repair the roof, remove the ash (WHC 1993), and operate the filtration system in a manner that will remove suspended particles. Changes in pH are also not a significant problem, since adverse effects from out of limit water pH are slow to occur and time would be available to formulate and take corrective actions.

The frequency of an ash fall event is estimated to be 2.08×10^{-3} per year (Herborn 1993).

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13.0 REFERENCES

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**ATTACHMENT
UNIT DOSES AND ATMOSPHERIC
DISPERSION COEFFICIENTS
FOR K-BASIN ISB**

August 16, 1994

Westinghouse Hanford Company

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1.0 PURPOSE

The purpose of this document is to provide technical information in support of the K basin Interim Safety Basis (ISB). Section 2.0 contains unit dose estimates for K basin sludge, water and fuel. Section 3.0 provides atmospheric dispersion coefficients (X/Q') for transport of the released material to downwind receptors. Section 4.0 discusses the use of the unit doses and X/Q' to calculate radiological doses for hypothetical accident scenarios developed elsewhere.

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2.0 RADIOLOGICAL DOSES

This section provides unit dose estimates for K basin sludge, water and fuel. Sludge and water doses are based on laboratory sample results. Fuel doses are based on theoretical estimates which account for fuel type, power histories and decay periods. Both inhalation and submersion pathways are considered. In all cases the submersion pathway is insignificant compared to the inhalation pathway and is therefore not reported separately. The basic assumptions made in the dose estimates are that the released material is 100% respirable. Plutonium is assumed to be in oxide form. Effective dose equivalent (EDE) and the limiting organ dose is reported for a 50 year commitment period for an acute uptake. Units are in rem per milli-liter (ml) or gram (g) inhaled.

2.1 KE BASIN

This section provides radiological dose results for laboratory samples of K East (KE) basin sludge and water samples. Dose units are in rem and apply to inhaled quantities. Although submersion doses are included, they comprise an insignificant contribution to the total.

To convert the dose per unit quantity inhaled to a dose per unit quantity released, the atmospheric dispersion coefficient and breathing rate must be estimated. These parameters are provided in Sections 3 and 4.

2.1.1 KE BASIN SLUDGE

Table 2-1. Radiological Analysis of KE Basin Sludge Samples - July 1993

Nitric Acid Digestion Results ($\mu\text{Ci/g}$ dried)	Sample ID			
	S3-032-01	S3-032-02	S3-032-03	S3-032-04 ¹
^{239,240} Pu ²	6.83	4.24	5.87	2.35
²⁴¹ Am ³	8.05	4.90	8.13	2.03
¹³⁷ Cs	42.8	47.5	41.5	13.1
⁹⁰ Sr	88.1	30.0	74.8	7.01
Residue Concentration as-settled (g/ml)	0.764	0.811	0.198	0.114
Sludge Density as-settled (g/ml)	1.53	1.54	1.13	1.11
Committed EDE per Unit Intake as-settled ⁴ (rem/ml)	4.3E+3	2.8E+3	1.1E+3	1.8E+2
Limiting Committed Organ Dose ⁵ as-settled ⁴ (rem/ml)	6.6E+4	4.3E+4	1.7E+4	2.6E+3

It is recommended that 1.0E+4 rem EDE/ml as-settled sludge be used as a safety basis to bound the variation found in these four sludge samples and to bound the additional dose contribution due to isotopes not analyzed by the laboratory (see footnote 2). Use of the recommended value and application to acceptance criteria placed on the EDE also ensures that the corresponding organ dose criteria, 10 times the EDE criteria (WHC 1992), is not exceeded.

¹Based on mean value.

²Although the isotopes of ²³⁸Pu, ²⁴¹Pu and ²⁴²Pu are not evaluated, these isotopes contribute 20% or less to the dose attributed to ²³⁹Pu, ²⁴⁰Pu and ²⁴¹Am in typical fuels and weapons grade mixtures.

³Activity concentration based on GEA (Gamma Energy Analysis)

⁴Code calculated doses converted from g dried to as-settled ml basis.

⁵Maximum organ was bone surface in all cases.

Use of the recommended value makes further evaluation of organ doses unnecessary.

Table 2-2. Radiological Analysis of KE Basin Sand Filter Backwash Pit (SFBWP) Samples - First Campaign - September 1993

Nitric Acid Digestion Results ($\mu\text{Ci/g}$ dried)	Sample ID		
	S3-059-03	S3-059-04	S3-059-11
$^{239,240}\text{Pu}$	7.55	3.51	2.25
^{238}Pu	NA ⁷	NA	0.39
^{241}Am	6.98	2.92	1.89 ²
^{137}Cs	NA	16.1	48.8
Total Beta	63.2	39.0	156
^{90}Sr ⁶	NA	11.5	54
Residue Concentration as-settled (g/ml)	0.123	0.483	0.452
Sludge Density as-settled (g/ml)	1.08	1.30	1.30
Committed EDE per Unit Intake as-settled ⁴ (rem/ml)	6.6E+2	1.2E+3	7.7E+4
Limiting Committed Organ Dose ⁵ as-settled ⁴ (rem/ml)	1.0E+3	1.6E+4	1.1E+4

⁶The laboratory did not specifically analyze sludge samples for ^{90}Sr . However, since the dominant beta contributors are ^{137}Cs , ^{90}Sr and ^{90}Y , the ^{90}Sr activity is estimated to be one-half of the total activity less the ^{137}Cs activity.

⁷Not analyzed.

Table 2-3. Radiological Analysis of KE SFBWP Transfer Channel Samples - First Campaign - September 1993

Nitric Acid Digestion Results ($\mu\text{Ci/g}$ dried)	Sample ID			
	S3-059-05	S3-059-06	S3-059-07	S3-059-08
$^{239,240}\text{Pu}$	45.7	5.02	3.00	4.41
^{241}Am	34.6	3.92	2.25	2.87
^{137}Cs	109	16.7	25.8	26.2
Total Beta	525	75.1	53.6	59.4
$^{90}\text{Sr}^6$	208	29.2	13.9	16.6
Residue Concentration as-settled (g/ml)	0.049	0.251	0.484	0.406
Sludge Density as-settled (g/ml)	1.08	1.21	1.41	1.31
Committed EDE per Unit Intake as-settled ⁶ (rem/ml)	1.5E+3	8.3E+2	9.2E+2	1.1E+3
Limiting Committed Organ Dose ⁵ as-settled ⁴ (rem/ml)	2.1E+4	1.2E+4	1.4E+4	1.5E+4

Table 2-4. Radiological Analysis of KE SFBWP - Second Campaign - November 1994

Nitric Acid Digestion Results ($\mu\text{Ci/g}$ dried)	Sample ID			
	S3-062-02	S3-062-03	S3-062-04	S3-062-05
$^{239,240}\text{Pu}$	3.01	2.18	5.85	2.20
^{238}Pu	NA	0.38	0.99	0.31
$^{241}\text{Am}^2$	2.70	1.83	4.56	1.46
^{137}Cs	32.1	14.7	43.4	15.1
Total Beta	98.0	57.2	125	34.9
$^{90}\text{Sr}^6$	33.0	21.3	40.8	9.9
Residue Concentration as-settled (g/ml)	0.308	0.479	0.288	0.594
Sludge Density as-settled (g/ml)	1.18	1.24	1.25	1.48
Committed EDE per Unit Intake as-settled ⁴ (rem/ml)	6.5E+2	7.7E+2	1.2E+3	8.3E+2
Limiting Committed Organ Dose ⁵ as-settled ⁴ (rem/ml)	9.5E+3	1.1E+4	1.7E+4	1.2E+4

Table 2-4 Cont. Radiological Analysis of KE SFBWP - Second Campaign - November 1994

Nitric Acid Digestion Results ($\mu\text{Ci/g}$ dried)	Sample ID		
	S3-062-06	S3-062-07	S3-062-08
$^{239,240}\text{Pu}$	0.94	33.7	3.95
^{238}Pu	NA	4.55	0.64
$^{241}\text{Am}^3$	0.81	30.2	3.17
^{137}Cs	12.2	119	13.0
Total Beta	21.0	313	40.9
$^{90}\text{Sr}^6$	4.4	97	14.0
Residue Concentration as-settled (g/ml)	0.818	0.056	0.386
Sludge Density as-settled (g/ml)	1.53	1.05	1.36
Committed EDE per Unit Intake as-settled ⁴ (rem/ml)	6.0E+2	1.4E+3	1.1E+3
Limiting Committed Organ Dose ⁵ as-settled ⁴ (rem/ml)	8.2E+3	2.0E+4	1.5E+4

It is recommended that 3E+3 rem EDE/ml as-settled sludge be used as a safety basis for releases involving SFBWP and transfer channel sludge. See discussion in last paragraph of Section 2.1.1 for further explanation.

2.1.2 KE BASIN WATER

The unit dose results listed in Table 2-4A are based on the laboratory sample data reported in Meichle (1993).

Table 2-4A. Radiological Analysis of KE Basin Water

Radionuclide	Inventory ($\mu\text{Ci/L}$)
$^{239,240}\text{Pu}$	3.3E-1
^{238}Pu	6.9E-2
^{241}Am	7.0E-3
^3H	3.4
^{90}Sr	2.5
^{137}Cs	6.1
Committed EDE per Unit Intake (rem/L)	1.2E+2
Limiting Committed Organ Dose (rem/L)	1.3E+3

2.1.3 KE BASIN FUEL

The KE basin fuel is subdivided into three groups for analysis purposes. These are total basin, peak canister and peak element. The total mass of fuel contained in each of these fuel groups is given in Table 2-6.

Table 2-6. KE Basin Fuel Inventories Used in ORIGEN2 Analyses

KEast Basin Total:	1146.16	MT
KEast Maximum MKIV Canister:	333.62	kg
KEast Maximum MKIV Element:	23.830	kg

Each of these fuel groups were analyzed using the ORIGEN2 computer code and the information provided in the SCATS (Safeguards Control and Accountability Transactions System) material accountability database (Wittekind, 1994). Only those isotopes contributing more than 0.1% to the inhalation dose are reported. Additionally, ^3H and ^{85}Kr are reported since the release fraction and exposure pathway can be much different for these isotopes.

The ORIGEN2 computer code was used to project radioactive decay and ^{241}Am build-up from 1995 to the year 2045 in five year increments. Since the maximum dose is achieved in approximately 50 years, radioisotopic inventories are provided for both 1/1/1995 and 1/1/2045.

Table 2-7. Radiological Analysis of KE Basin Fuel - Total Basin Inventory

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
^{238}Pu	5.173E+04	3.489E+04
^{239}Pu	1.120E+05	1.118E+05
^{240}Pu	5.993E+04	5.963E+04
^{241}Pu	2.945E+06	2.653E+05
^{241}Am	1.612E+05	2.333E+05
^{244}Cm	6.546E+03	9.658E+02
^3H	1.612E+04	9.737E+02
^{85}Kr	2.730E+05	1.077E+04
^{90}Sr	4.537E+06	1.380E+06
^{90}Y	4.538E+06	1.380E+06
^{137}Cs	5.830E+06	1.836E+06
Committed EDE per Unit Intake (rem/g)	1.3E+5	1.4E+5
Limiting Committed Organ Dose ⁵ (rem/g)	1.9E+6	2.2E+6

Table 2-8. Radiological Analysis of KE Basin Fuel - Maximum MKIV Canister

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	3.983E+01	2.688E+01
²³⁹ Pu	4.806E+01	4.799E+01
²⁴⁰ Pu	3.580E+01	3.564E+01
²⁴¹ Pu	2.048E+03	1.846E+02
²⁴¹ Am	1.166E+02	1.664E+02
²⁴⁴ Cm	8.922E+00	1.316E+00
³ H	7.860E+00	4.749E-01
⁸⁵ Kr	1.205E+02	4.752E+00
⁹⁰ Sr	2.136E+03	6.497E+02
⁹⁰ Y	2.136E+03	6.499E+02
¹³⁷ Cs	2.911E+03	9.168E+02
Committed EDE per Unit Intake (rem/g)	3.1E+5	3.3E+5
Limiting Committed Organ Dose ⁵ (rem/g)	4.6E+6	5.2E+6

Table 2-9. Radiological Analysis of KE Basin Fuel - Maximum MKIV Element

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	1.499E+01	1.011E+01
²³⁹ Pu	4.947E+00	4.940E+00
²⁴⁰ Pu	6.277E+00	6.298E+00
²⁴¹ Pu	4.854E+02	4.373E+01
²⁴¹ Am	2.754E+01	3.935E+01
²⁴⁴ Cm	2.281E+01	3.365E+00
³ H	1.338E+00	8.083E-02
⁸⁵ Kr	1.722E+01	6.793E-01
⁹⁰ Sr	2.986E+02	9.084E+01
⁹⁰ Y	2.987E+02	9.086E+01
¹³⁷ Cs	4.754E+02	1.497E+02
Committed EDE per Unit Intake (rem/g)	1.2E+6	1.0E+6
Limiting Committed Organ Dose ⁵ (rem/g)	1.8E+7	1.6E+7

2.2 KW BASIN

2.2.1 KW BASIN SLUDGE

This data not yet available.

2.2.2 KW BASIN WATER

This data not yet available.

2.2.3 KW BASIN FUEL

The KW basin fuel is subdivided into five groups. These are total basin, peak canister containing Mark IA fuel, peak canister containing Mark IV fuel, peak element containing Mark IA fuel, and peak element containing Mark IV fuel. The total mass of fuel contained in each of these fuel groups is given in Table 2-10. The ORIGEN2 analysis of KW basin fuel inventories similarly followed the approach used for the KE basin.

Table 2-10. KW Basin Fuel Inventories Used in ORIGEN2 Analyses

KWest Basin Total:	952.198	MT
KWest Maximum MKIV Canister:	333.62	kg
KWest Maximum MKIV Element:	23.830	kg
KWest Maximum MKIA Canister:	235.796	kg
KWest Maximum MKIA Element:	16.8426	kg

Table 2-11. Radiological Analysis of KW Basin Fuel - Total Basin Inventory

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	4.528E+04	3.053E+04
²³⁹ Pu	1.044E+05	1.042E+05
²⁴⁰ Pu	5.419E+04	5.391E+04
²⁴¹ Pu	2.849E+06	2.567E+05
²⁴¹ Am	1.315E+05	2.031E+05
²⁴⁴ Cm	3.851E+03	5.681E+02
³ H	1.719E+04	1.039E+03
⁸⁵ Kr	2.998E+05	1.182E+04
⁹⁰ Sr	4.806E+06	1.462E+06
⁹⁰ Y	4.808E+06	1.462E+06
¹³⁷ Cs	6.043E+06	1.903E+06
Committed EDE per Unit Intake (rem/g)	1.4E+5	1.5E+5
Limiting Committed Organ Dose ⁵ (rem/g)	2.0E+6	2.3E+6

Table 2-12. Radiological Analysis of KW Basin Fuel - Maximum MKIA Canister

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	1.463E+01	9.867E+00
²³⁹ Pu	2.800E+01	2.796E+01
²⁴⁰ Pu	1.694E+01	1.685E+01
²⁴¹ Pu	8.567E+02	7.718E+01
²⁴¹ Am	4.502E+01	6.613E+01
²⁴⁴ Cm	1.437E+00	2.120E-01
³ H	4.821E+00	2.913E-01
⁸⁵ Kr	8.335E+01	3.287E+00
⁹⁰ Sr	1.429E+03	4.347E+02
⁹⁰ Y	1.430E+03	4.348E+02
¹³⁷ Cs	1.802E+03	5.676E+02
Committed EDE per Unit Intake (rem/g)	1.8E+5	2.0E+5
Limiting Committed Organ Dose ⁵ (rem/g)	2.6E+6	3.0E+6

Table 2-13. Radiological Analysis of KW Basin Fuel - Maximum MKIA Element

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	4.726E+00	3.187E+00
²³⁹ Pu	2.838E+00	2.834E+00
²⁴⁰ Pu	2.962E+00	2.953E+00
²⁴¹ Pu	2.104E+02	1.896E+01
²⁴¹ Am	1.104E+01	1.623E+01
²⁴⁴ Cm	2.771E+00	4.088E-01
³ H	2.324E-02	1.404E-03
⁸⁵ Kr	1.123E+01	4.431E-01
⁹⁰ Sr	1.894E+02	5.763E+01
⁹⁰ Y	1.895E+02	5.764E+01
¹³⁷ Cs	2.644E+02	8.329E+01
Committed EDE per Unit Intake (rem/g)	5.8E+5	5.9E+5
Limiting Committed Organ Dose ⁵ (rem/g)	8.7E+6	9.5E+6

Table 2-14. Radiological Analysis of KW Basin Fuel - Maximum MKIV Canister

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	3.754E+01	2.533E+01
²³⁹ Pu	4.712E+01	4.705E+01
²⁴⁰ Pu	3.415E+01	3.399E+01
²⁴¹ Pu	2.200E+03	1.982E+02
²⁴¹ Am	1.006E+02	1.560E+02
²⁴⁴ Cm	8.373E+00	1.235E+00
³ H	8.844E+00	5.343E-01
⁸⁵ Kr	1.399E+02	5.518E+00
⁹⁰ Sr	2.211E+03	6.727E+02
⁹⁰ Y	2.212E+03	6.728E+02
¹³⁷ Cs	2.991E+03	9.420E+02
Committed EDE per Unit Intake (rem/g)	2.8E+5	3.1E+5
Limiting Committed Organ Dose ⁵ (rem/g)	4.0E+6	4.9E+6

Table 2-15. Radiological Analysis of KW Basin Fuel - Maximum MKIV Element

Radionuclide	Inventory 1/1/1995 (Ci)	Inventory 1/1/2045 (Ci)
²³⁸ Pu	1.390E+01	9.377E+00
²³⁹ Pu	4.882E+00	4.875E+00
²⁴⁰ Pu	6.029E+00	6.047E+00
²⁴¹ Pu	5.251E+02	4.731E+01
²⁴¹ Am	2.392E+01	3.715E+01
²⁴⁴ Cm	2.119E+01	3.126E+00
³ H	1.489E+00	8.993E-02
⁸⁵ Kr	1.988E+01	7.842E-01
⁹⁰ Sr	3.076E+02	9.357E+01
⁹⁰ Y	3.077E+02	9.360E+01
¹³⁷ Cs	4.833E+02	1.522E+02
Committed EDE per Unit Intake (rem/g)	1.1E+6	1.0E+6
Limiting Committed Organ Dose ⁵ (rem/g)	1.6E+7	1.6E+7

2.3 RESULTS SUMMARY

There is a broad spectrum of hazard in terms of dose for the various radioactive materials contained in the K basins. For accident analysis purposes, it is usually necessary to choose a material which bounds the possibilities. Table 2-16 provides recommended unit doses for the radioactive materials analyzed in Section 2. Additionally, unit doses for radioactive cesium and strontium elements and their daughters are included. Units are in rem EDE per unit quantity inhaled. These recommended values provide a safety basis which is both simplified and conservatively bounding. Use of these values should ensure that calculated consequences are maximized with respect to decay time, power history, and sample variation. Use of the recommended unit doses as an effective dose equivalent (EDE) also ensures that organ dose criteria are satisfied as well (i.e. the user does not need to calculate organ doses).

Table 2-16. Radiological Analysis Results Summary (rem/unit-inhaled)

KE Sludge:	
Basin	1.0E+4 rem EDE/ml as-settled sludge
SFBWP and Transfer Channel	3.0E+3 " " " "
KE Water:	
	1.3E+2 rem EDE/L
KE Fuel: ⁸	
Total Basin	Use KW value
Maximum MKIV Canister	5.2E+5 rem EDE/g
Maximum MKIV Element	1.8E+6 rem EDE/g
KW Sludge:	
Basin	NA ⁹
SFBWP and Transfer Channel	NA
KW Water:	
	NA
KW Fuel: ⁸	
Total Basin	2.3E+5 rem EDE/g
Maximum MKIA Canister	3.0E+5 rem EDE/g
Maximum MKIA Element	9.5E+5 rem EDE/g
Maximum MKIV Canister	Use KE value
Maximum MKIV Element	Use KE value
⁹⁰ Sr/ ⁹⁰ Y ¹⁰	2.7E+5 rem/Ci
¹³⁷ Cs/ ^{137m} Ba ¹⁰	3.2E+4 rem/Ci

⁸Transuranic elements account for 99.5% or more of EDE.

⁹Not available.

¹⁰Daughter product assumed to be in transient equilibrium.

3.0 ATMOSPHERIC TRANSPORT

Atmospheric transport calculations provide estimates of air concentration resulting from atmospheric discharges of radionuclides. These air concentrations are then used to calculate doses to man. The following sections discuss the models and data used to perform these calculations.

3.1 RELEASE POINT TO RECEPTOR DISTANCES

Site and river boundary distances used in this report are based on the 105-KW storage facility location. Atmospheric transport analysis based on the location of 105-KW conservatively bounds those that might use the 105-KE location. The difference is not great. Both onsite and site boundary receptors are considered. Additionally, receptors located on the near bank of the Columbia River are also considered. The sector yielding the maximum dispersion coefficient is chosen as the bounding value to be used in safety analysis. The receptor is assumed to remain at the given location for the duration of plume passage. ~~The maximum onsite receptor is assumed to be at a distance of 100 m~~ The maximum onsite receptor is assumed to be the Fire Station at a distance of 4000 m ESE, based on the information in Table 3-3. Hanford site boundary distances are given in Tables 3-1 and 3-2.

Table 3-1. Distance from 105-KW to 1994 Hanford Site Boundary.

Transport direction	Distance (km)	Transport direction	Distance (km)
S	25.15	N	10.91
SSW	25.54	NNE	15.19
SW	17.31	NE	19.88
WSW	15.66	ENE	20.38
W	12.04	E	20.57
WNW	10.07	ESE	21.70
NW	8.65	SE	30.94
NNW	8.81	SSE	30.99

Table 3-2. Distance from 105-KW to Near Bank of Columbia River.

Transport direction	Distance (km)	Transport direction	Distance (km)
S	N/A	N	0.53
SSW	N/A	NNE	0.73
SW	N/A	NE	2.25
WSW	1.06	ENE	11.33
W	0.71	E	11.78
WNW	0.52	ESE	15.05
NW	0.48	SE	N/A
NNW	0.48	SSE	N/A

3.2 FREQUENCY-BASED X/Q'

The atmospheric dispersion coefficient X/Q' represents the dilution of an airborne contaminant due to atmospheric mixing and turbulence. It is the ratio of the average contaminant air concentration X at the receptor to the contaminant release rate Q' at the release point. Since atmospheric conditions fluctuate, a bounding atmospheric condition is determined to be that condition which causes a downwind concentration of airborne contaminants that is exceeded only a small fraction of time due to weather fluctuations. The Nuclear Regulatory Commission (NRC) defines this fraction as 0.5% for each sector or 5% for the overall Site, whichever is greater (NRC 1982). Bounding X/Q' values are provided in this report.

3.3 GAUSSIAN STRAIGHT LINE PLUME

Not only is the straight-line Gaussian plume model the most widely used method of estimating downwind air concentrations of radionuclides released to the atmosphere, but also it is the most often verified atmospheric dispersion model (Miller 1984). For a continuous point source and invariant meteorology, this model is given by

$$\frac{\bar{X}}{Q'} = \frac{f(y)g(z)}{2\pi\sigma_y\sigma_z\bar{u}} \quad (1)$$

where:

- σ_y = Horizontal diffusion coefficient (m)
- σ_z = Vertical diffusion coefficient (m)
- $f(y)$ = Horizontal correction term
- $g(z)$ = Vertical correction term
- $\bar{u}$ = Mean wind speed (m/s).

The horizontal correction term is given by:

$$f(y) = \exp \left[-\frac{1}{2} \left(\frac{y}{\sigma_y} \right)^2 \right] \quad (2)$$

where:

y = Horizontal offset from plume centerline (m).

If the mixing height is large compared to the release height and the vertical diffusion coefficient, then the vertical correction term is given by:

$$g(z) = F_{\text{ref}} \exp \left[-\frac{1}{2} \left(\frac{h_e - z}{\sigma_z} \right)^2 \right] \quad (3)$$

where:

F_{ref} = Reflection factor¹¹
 h_e = Effective release height (m)
 z = Receptor height (m).

Mathematical approximations are used to evaluate the atmospheric diffusion coefficients σ_y & σ_z which are based on the Pasquill-Gifford-Turner curves. These approximations are identical to those used in the GENII Code (Napier 1988).

¹¹The reflection factor is based on the virtual image source and assumes 100% ground and mixing layer reflection. The mathematical expression is given in Hey (1993a).

3.4 PLUME MEANDER

To account for the increased dispersion due to a fluctuating wind direction over an extended release period, a plume meander model based on the empirical correlation recommended by the NRC (1982) is used. When selected, it effectively increases the value of the horizontal diffusion coefficient. The amount of increase is dependent upon both atmospheric stability class and wind speed. The procedure is given below.

For $1 \text{ m/s} \leq u \leq 2 \text{ m/s}$:

M =	6	Stability Class	G
	4		F
	3		E
	2		D

For $2 \text{ m/s} < u \leq 6 \text{ m/s}$:

M =	$(u/6)^{(-\ln(6)/\ln(3))}$	Stability Class	G
	$(u/6)^{(-\ln(4)/\ln(3))}$		F
	$(u/6)^{(-\ln(3)/\ln(3))}$		E
	$(u/6)^{(-\ln(2)/\ln(3))}$		D

For $x < 800 \text{ m}$:

$$\Sigma_y = M \sigma_y(x) \quad (4)$$

For $x \geq 800 \text{ m}$:

$$\Sigma_y = (M-1) \sigma_y(800) + \sigma_y(x) \quad (5)$$

3.5 BUILDING WAKE

The dispersion of a plume can also be enhanced by the added turbulence of a building wake. To account for this effect, a building wake model taken from NRC (1990) is employed. The model does not alter the Gaussian shape of the plume; but initializes the width and height of the plume on the basis of the building dimensions. It assumes the release point to be at the ground level and in the middle of the downwind face of the building. The plume horizontal and vertical dimensions are initialized such that the plume concentrations at the sides and roof line of the building from which the release occurs are 10% of plume centerline concentrations (i.e., building edges are 2.15σ from the plume centerline). The formulae below were taken from Equation 2.2 of NRC (1990).

$$\sigma_y(x=0) = \frac{W_b}{4.3} \quad (6)$$

where:

W_b = Building width (m).

$$\sigma_z(x=0) = \frac{H_b}{2.15} \quad (7)$$

where:

H_b = Building height (m).

The above adjustment is made using the virtual distance method. The virtual distance method involves calculating the distances corresponding to $\sigma_y(x=0)$ and $\sigma_z(x=0)$ and adding these distances to the distance from the release point to receptor (x). These adjusted "virtual distances" are then used to calculate subsequent diffusion coefficients.

3.6 ATMOSPHERIC DISPERSION COEFFICIENTS

A 99.5% sector and 95% overall site X/Q' (NRC 1982), suitable for safety analyses, are calculated for various site locations as well as for the near river bank and site boundary receptors assuming ground level releases. The greater of the two is the 99.5% sector X/Q' in all cases. The effects of plume meander and building wake, as discussed in previous sections are also evaluated. Building wake is based on a conservatively small building dimension of 67 ft (20 m) wide and 16 ft (5 m) tall (Meichle 1993). Results are provided in Table 3-3.

Table 3-3. Maximum 99.5% Sector Atmospheric Dispersion Coefficients

Receptor/Scenario Description	X/Q' (s/m ³)			
	Point Source	W/ Plume Meander	W/ Building Wake	W/ PM and BW
100m Radius (100m E)	7.32E-2	1.24E-2	1.40E-2	5.67E-3
Near River Bank (480m NW)	2.15E-3	5.55E-4	1.53E-3 ¹²	4.54E-4
N Reactor (3200m NE)	1.13E-4	6.13E-5	1.07E-4	6.01E-5
Fire Station (4000m ESE)	1.47E-4	7.37E-5	1.40E-4	7.24E-5
B & C Reactor (4000m WSW)	8.51E-5	5.02E-5	8.20E-5	4.96E-5
1994 Hanford Site Boundary (12040m W)	3.58E-5	2.60E-5	same as point source	same as w/ PM

The title of the joint frequency file used in the computation of the X/Q' is:

100 N AREA - 10 M - Pasquill A - G (1983 - 1991 Average)
Created 8/26/92 KR

The GXQ (Hey 1993a and b) output files for these calculations are provided in Appendix A.

¹²Maximum receptor located 710m W.

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4.0 DOSE CONSEQUENCE ESTIMATION

The atmospheric dispersion coefficients and unit doses provided in this report can be used to determine the effective dose equivalent and the committed dose to the limiting organ. This is typically done using the following equation.

$$D \text{ (rem)} = Q \text{ (ml)} \times \frac{\lambda}{Q'} \text{ (s/m}^3\text{)} \times R \text{ (m}^3\text{/s)} \times C \text{ (rem/ml)} \quad (8)$$

where:

- Q = Quantity respirable radioactive material released (g or ml)
- λ/Q' = Time integrated atmospheric dispersion coefficient (s/m^3)
- R = $3.3\text{E-}4 \text{ m}^3\text{/s}$ standard man acute breathing rate (ICRP 1975)
- C = Dose per unit respirable radioactive material inhaled (rem/(g or ml))

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APPENDIX A

GXQ Output Files

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GXQ Version 3.1 D
June 8, 1993

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

Users Guide documented in WHC-SD-GN-SWD-3002 Rev. 0.
Validation documented in WHC-SD-GN-SWD-3003 Rev. 0.
Code Custodian is Brit E. Hey, WHC, ext. 6-2921.

Run Date = 9/27/1994
Run Time = 8:56 4.97

INPUT ECHO:

105-KW X/Q Calculation w/ Building Wake and Plume Meander

c

c MODE CHOICE:

c mode = 1 then X/Q based on Hanford site specific meteorology

c mode = 2 then X/Q based on atmospheric stability class and wind speed

c mode = 3 then X/Q plot file is created

c mode

1

c

c LOGICAL CHOICES:

c ifox inorm icdf ichk isite ipop icon

T F F F F F F
c ifox = t then joint frequency used to compute frequency to exceed X/Q

c = f then joint frequency used to compute annual average X/Q

c inorm = t then joint frequency data is normalized (as in GENII)

c = f then joint frequency data is un-normalized

c icdf = t then cumulative distribution file created (CDF.OUT)

c = f then no cumulative distribution file created

c ichk = t then X/Q parameter print option turned on
 c = f then no parameter print
 c isite = t then X/Q based on joint frequency data for all 16 sectors
 c = f then X/Q based on joint frequency data of individual sectors
 c ipop = t then X/Q is population weighted
 c = f then no population weighting
 c icon = t then X/Q is air concentration
 c = f then X/Q is integrated exposure

c

c MODEL CHOICES:

c idep iwake ipm irise igrav iwash iflow iwind
 0 2 1 0 0 0 0 0
 c idep = 1 then plume depletion model turned on (Chamberlain model)
 c iwake = 1 then NRC RG 1.145 building wake model turned on
 c = 2 then MACCS virtual distance building wake model turned on
 c ipm = 1 then NRC RG 1.145 plume meander model turned on
 c = 2 then 5th Power Law plume meander model turned on
 c = 3 then sector average model turned on
 c irise = 1 then momentum/buoyancy plume rise model turned on, buoyancy
 c rise based on sensible heat emission
 c = 2 then momentum/buoyancy plume rise model turned on, buoyancy
 c rise based on initial plume density
 c igrav = 1 then gravitational settling model turned on
 c iwash = 1 then stack downwash model turned on
 c iflow = 1 then sigmas adjusted for volume flow rate
 c iwind = 1 then wind speed corrected for plume height
 c = 0 to turn any of the above models off
 c

c PARAMETER INPUT:

c	stack	wind		frequency	
c	release	speed	mixing	to	scaling
c	height	height	height	exceed	factor
c	(m)	(m)	(m)	(%)	(?)
	0.00000E+00	1.00000E+01	1.00000E+03	5.00000E-01	1.00000E+00
c					
c					gravitational
c	building	building	release	deposition	settling
c	width	height	duration	velocity	velocity
c	(m)	(m)	(hr)	(m/s)	(m/s)
	2.00000E+01	5.00000E+00	1.00000E+00	1.00000E-03	1.00000E-02
c					
c		initial		sensible	
c	initial	plume		heat	
c	plume	flow	stack	emission	
c	density	rate	diameter	rate	
c	(g/cc)	(m3/s)	(m)	(cal/s)	
	3.10000E-04	7.07000E+00	3.00000E+00	2.80000E+06	

c RECEPTOR DEPENDENT DATA

MODE:

Site specific X/Q calculated

LOGICAL CHOICES:

Joint frequency used to calculate X/Q based on frequency of exceedance.

No normalization of joint frequency.

X/Q calculated for single sector.

Output is atmospheric dispersion coefficient.

MODELS SELECTED:

MACCS Virtual source building wake

NRC RG 1.145 plume meander

WARNING/ERROR MESSAGES:

JOINT FREQUENCY DATA:

100 N AREA - 10 M - Pasquill A - G (1983 - 1991 Average)

Created 8/26/92 KR

105-KW. X/Q Calculation w/ Building Wake and Plume Meander

SECTOR	DISTANCE (m)	RECEPT HEIGHT (m)	SECT. FREQ. (%)	POPULATION	TOTAL POPULATION SCALED X/Q (s/m3)	AVERAGE INDIVIDUAL SCALED X/Q (s/m3)	ATM. STAB. CLASS	WIND SPEED (m/s)
S	100	0	6.18	1	4.87E-03	4.87E-03	F	.89
SSW	100	0	5.01	1	4.06E-03	4.06E-03	F	.89
SW	100	0	3.91	1	4.37E-03	4.37E-03	F	.89
WSW	100	0	3.88	1	4.62E-03	4.62E-03	F	.89
W	100	0	7.54	1	5.34E-03	5.34E-03	F	.89
WNW	100	0	5.46	1	4.61E-03	4.61E-03	F	.89
NW	100	0	4.92	1	4.35E-03	4.35E-03	F	.89
NNW	100	0	3.41	1	3.65E-03	3.65E-03	E	.89
N	100	0	3.89	1	4.21E-03	4.21E-03	F	.89
NNE	100	0	3.54	1	3.85E-03	3.85E-03	E	.89
NE	100	0	5.96	1	4.52E-03	4.52E-03	F	.89
ENE	100	0	9.19	1	5.17E-03	5.17E-03	F	.89
E	100	0	16.14	1	5.67E-03	5.67E-03	G	.89
ESE	100	0	10.16	1	5.31E-03	5.31E-03	F	.89
SE	100	0	6.17	1	5.06E-03	5.06E-03	F	.89
SSE	100	0	4.44	1	4.89E-03	4.89E-03	F	.89
NE	3200	0	5.96	1	6.01E-05	6.01E-05	F	.89
ESE	4000	0	10.16	1	7.24E-05	7.24E-05	F	.89
WSW	4000	0	3.88	1	4.96E-05	4.96E-05	F	.89

S	25150	0	6.18	1	8.15E-06	8.15E-06 F	.89
SSW	25540	0	5.01	1	6.19E-06	6.19E-06 G	2.65
SW	17310	0	3.91	1	1.03E-05	1.03E-05 F	.89
WSW	15660	0	3.88	1	1.21E-05	1.21E-05 F	.89
W	12040	0	7.54	1	2.59E-05	2.59E-05 F	.89
WNW	10070	0	5.46	1	1.92E-05	1.92E-05 F	.89
NW	8650	0	4.92	1	2.14E-05	2.14E-05 G	2.65
NNW	8810	0	3.41	1	1.24E-05	1.24E-05 G	4.70
N	10910	0	3.89	1	1.59E-05	1.59E-05 G	2.65
NNE	15190	0	3.54	1	6.95E-06	6.95E-06 G	4.70
NE	19880	0	5.96	1	9.07E-06	9.07E-06 F	.89
ENE	20380	0	9.19	1	1.33E-05	1.33E-05 F	.89
E	20570	0	16.14	1	1.82E-05	1.82E-05 G	.89
ESE	21700	0	10.16	1	1.37E-05	1.37E-05 F	.89
SE	30940	0	6.17	1	7.73E-06	7.73E-06 F	.89
SSE	30990	0	4.44	1	6.58E-06	6.58E-06 F	.89
S	25150	0	6.18	1	8.15E-06	8.15E-06 F	.89
SSW	25540	0	5.01	1	6.19E-06	6.19E-06 G	2.65
SW	17310	0	3.91	1	1.03E-05	1.03E-05 F	.89
WSW	1060	0	3.88	1	1.72E-04	1.72E-04 F	.89
W	710	0	7.54	1	3.11E-04	3.11E-04 G	4.70
WNW	520	0	5.46	1	4.36E-04	4.36E-04 F	.89
NW	480	0	4.92	1	4.54E-04	4.54E-04 G	2.65
NNW	480	0	3.41	1	3.64E-04	3.64E-04 G	2.65
N	530	0	3.89	1	3.63E-04	3.63E-04 G	2.65
NNE	730	0	3.54	1	2.07E-04	2.07E-04 G	2.65
NE	2250	0	5.96	1	8.37E-05	8.37E-05 F	.89

ENE	11330	0	9.19	1	2.45E-05	2.45E-05 F	.89
E	11780	0	16.14	1	3.20E-05	3.20E-05 G	.89
ESE	15050	0	10.16	1	2.01E-05	2.01E-05 F	.89
SE	30940	0	6.17	1	7.73E-06	7.73E-06 F	.89
SSE	30990	0	4.44	1	6.58E-06	6.58E-06 F	.89

Execution Time = 0 hr 0 min 2.47 sec
 Stop - Program terminated.

APPENDIX B

GENII Input Files

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Program GENII Input File ##### 8 Jul 88 ####
 Title: Onsite, Inhalation & Submersion, Acute, Ground Level Release

OPTIONS===== Default =====
 F Near-field scenario? (Far-field) NEAR-FIELD: narrowly-focused
 F Population dose? (Individual) release, single site
 T Acute release? (Chronic) FAR-FIELD: wide-scale release,
 Maximum Individual data set used multiple sites
 Complete
 TRANSPORT OPTIONS===== Section EXPOSURE PATHWAY OPTIONS===== Section
 T Air Transport 1 F Finite plume, external 5
 F Surface Water Transport 2 T Infinite plume, external 5
 F Biotic Transport (near-field) 3,4 F Ground, external 5
 F Waste Form Degradation (near) 3,4 F Recreation, external 5
 T Inhalation uptake 5,6
 REPORT OPTIONS===== F Drinking water ingestion 7,8
 T Report AEDE only F Aquatic foods ingestion 7,8
 T Report by radionuclide F Terrestrial foods ingestion 7,9
 F Report by exposure pathway F Animal product ingestion 7,10
 F Debug report on screen F Inadvertent soil ingestion

INVENTORY #####

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

Use when	---Release Terms--- transport selected			-----Basic Concentrations----- near-field scenario, optionally				
	Air /yr	Surface Water /yr	Buried Waste /m3	Air /m3	Surface Soil /unit	Deep Soil /m3	Ground Water /L	Surface Water /L
Release								
Radio-								
nuclide								
PU238	3.1E+04							
PU239	1.0E+05							
PU240	5.4E+04							
PU241	2.6E+05							
AM241	2.0E+05							
CM244	5.7E+02							
H 3	1.0E+03							
KR85	1.2E+04							
SR90	1.5E+06							
Y 90	1.5E+06							
CS137	1.9E+06							

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

TIME #####

```

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

```

FAR-FIELD SCENARIOS (IF POPULATION DOSE)

```

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

```

NEAR-FIELD SCENARIOS

```

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

```

TRANSPORT

```

====AIR TRANSPORT=====SECTION 1=====
0 Calculate PM 0 Release type (0-3)
1 Option: 1-Use chi/Q or PM value F Stack release (T/F)
2-Select MI dist & dir 0 Stack height (m)
3-Specify MI dist & dir 0 Stack flow (m3/sec)
3.18E-6 Chi/Q or PM value 0 Stack radius (m)
0 MI sector index (1=S) 0 Effluent temp. (C)
0 MI distance from release point (m) 0 Building x-section (m2)
F Use jf data, (T/F) else chi/Q grid 0 Building height (m)

```

```

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

```

```

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

```

```

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

```


EXPOSURE #####

====EXTERNAL EXPOSURE=====SECTION 5=====

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

====INHALATION=====SECTION 6=====

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

====INGESTION POPULATION=====SECTION 7=====

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

Note below: S* or Source: 0-none, 1-ground water, 2-surface water
3-Derived concentration entered above

==== AQUATIC FOODS / DRINKING WATER INGESTION=====SECTION 8=====

F Salt water? (default is fresh)

USE ? FOOD T/F TYPE	TRAN- SIT hr	PROD- UCTION kg/yr	-CONSUMPTION- HOLDUP da	RATE kg/yr	DRINKING WATER
F FISH	0.00	0.0E+00	0.00	0.0	0 Source (see
F MOLLUS	0.00	0.0E+00	0.00	0.0	T Treatment? T/F
F CRUSTA	0.00	0.0E+00	0.00	0.0	0
F PLANTS	0.00	0.0E+00	0.00	0.0	0 Consumption

above)
Holdup/transit(da)
(L/yr)

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

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