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COMMENTS ON THE BRITISH NUCLEAR FUELS LIMITED, INC. DESIGN SAFETY FEATURES FEBRUARY 1999, RPT–W375–RU00001 Revision 0

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QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: CNWRA-generated original data contained in this report meet quality assurance requirements described in the CNWRA Quality Assurance Manual. Sources for other data should be consulted for determining the level of quality for those data.

ANALYSES AND CODES: No computer codes were used for analyses contained in this report.

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1 INTRODUCTION

The Tank Waste Remediation System-Privatization (TWRS-P) program was established in 1991 to manage the maintenance and clean up of radioactive wastes contained in 177 aging underground storage tanks. The TWRS-P contract is divided into two phases. Phase I, Part A, completed in 1998, consisted of a demonstration of technologies, development of conceptual design, and submittal of safety, regulatory licensing, and financial plans. Phase I, Part B, is divided into Parts B-1 and B-2. Part B-1 is effective from August 1998 through August 2000, after which time Part B-2 will commence. Phase II begins after Part B-2 is complete.

During Phase B-1, the TWRS-P contract requires British Nuclear Fuels Limited, Inc. (BNFL, Inc.) to provide a Design Safety Feature (DSF) deliverable at 6 mo from authorization. This deliverable was provided to the Regulatory Unit (RU) on February 24, 1999, for review. The deliverable consists of two volumes. Volume 1 contains a description of planned important to safety (ITS) structures, systems, and components (SSCs). In volume II, 10 representative examples were selected to demonstrate application of the Integrated Safety Management (ISM) process to systematically define ITS SSCs and DSFs.

Per Nuclear Regulatory Commission (NRC) guidance, the Center for Nuclear Waste Regulatory Analyses (CNWRA) review focused on volume II of the deliverable. Nine of the 10 examples were reviewed and as draft comments were generated, they were provided to the NRC by e-mail. This report provides a compilation of the CNWRA comments already submitted to the NRC.

2 GENERAL COMMENTS

2.1 REFERENCES

In each example, a considerable amount of key supporting information (e.g., reliability data, consequence calculations, selected conceptual models, modeling input parameter data, modeling assumptions, and event frequencies) is referenced to sources not available for review. In addition, little or no discussion of the applicability of a number of key references to the TWRS-P conditions is provided in the main report. The reader should be able to clearly (and efficiently) identify what information was obtained from referenced materials, what the rationale was for choosing referenced models, supporting data and input parameters; and why such information is relevant to TWRS-P analyses. For example, details of the consequence calculation for pretreatment pump drop, the text of the main report (page 3.3-6 and 7) cites a calculation by Smith.[1] The colocated worker and public dose calculation in Smith use a Chi/Q value that is referenced to in Kummerer.[2] Kummerer, however, was not provided in the reference materials. The reference is also similarly cited in CALC–W375HV–NS00004.[3] Such information, important to the dose calculations, should be available in the main report or at least in the enclosed reference material. Readers should not have to consult secondary references to identify important details of how a calculation was executed.

2.2 CHI/Q VALUES FOR CONSEQUENCE CALCULATIONS

Neither the main report nor the enclosed consequence (dose) calculations contain information on bases, models, and assumptions for the Chi/Q value (an important factor summarizing airborne dispersion effects for release consequence calculations). Bases for selected models and parameters that determined Chi/Q values and their relevance to known site conditions are not provided. This issue was identified initially in the following calculations: CALC–W375PT–SA00001,[4] CALC–W375–NS00003,[5] and CALC–W375HV–NS00004.[6]

2.3 CONSIDERATION OF UNCERTAINTIES

All the frequency and consequence calculations were deterministic with no quantitative analysis of uncertainties provided (e.g., stochastic calculations based on parameter distributions). Several frequency and consequence calculations were described as "upper bound." If this can be verified, the estimated frequencies

[1]Smith, D.A. 1999. *Consequences of Dropped Pump.* Preliminary Calculation W375–NS00003. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

[2]Kummerer, M. 1999. *Atmospheric Dispersion Coefficients for TWRS Accident Analysis.* CALC–W375–NS00005. Revision A. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

[3]Smith, D.A. 1999. *HLW Melter Feed Line Failure.* CALC–W375HV–NS00004. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

[4]Wojdac, L.F. 1999. *Loss of Cooling Water to Cesium Storage Tank, At Test Case.* Revision A. CALC–W375PT–SA00001. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

[5]See footnote 1.

[6]See footnote 3.

and consequences alleviate the need for estimation of uncertainties by bounding at the conservative end of the range.

2.4 INCONSISTENT APPLICATION OF ASSUMPTIONS

In Section 3.8, Receipt Vessel Rupture, it is assumed in the example that the high-level waste (HLW) receipt vessel is full to its overflow volume (235 m^3). This assumption is inconsistent with the example in Section 3.1, Hydrogen Generation in the HLW Storage Vessels, where the overflow volume for the same tank is assumed as 242 m^3.

2.5 INCONSISTENT SELECTION OF RADIONUCLIDES

The radionuclides selected for calculating consequences from receipt tank rupture as shown in table 2 of CALC–W375PT–NS00001[7] are different compared with radionuclides selected for calculating the hydrogen generation rate in table 2.0 of CALC–W375–NS00001.[8] In addition, both the dose magnitude and the unit are also different. The two inventories should be identical unless justification is provided for excluding radionuclides from one or the other of the calculations.

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[7]Kummerer, M. 1999. *Assignment of Severity Levels for Receipt Tank Rupture.* CALC–W375PT–NS00001. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

[8]Kummerer, M. 1999. *Assignment of Severity Levels for Hydrogen Deflagration Test Case.* CALC–W375–NS00001. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

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3 COMMENTS ON VOLUME II, SECTION 3.2, LOSS OF COOLING TO THE Cs STORAGE VESSEL

3.1 FAILURE OF COOLING COILS

BNFL, Inc. did not consider the frequency of occurrence of the failure of the cooling coil due to corrosion from the water side to be significant compared to the offsite power failure. In section 3.2.1.4, however, BNFL, Inc. indicated that detection of a radioactive leak in the cooling system may stop pump flow. Further, in section 3.5.2.6, BNFL, Inc. indicated that the frequency of cooling coil failure in the Sellafield waste processing facility due to water side corrosion is 1×10^{-3} to 0.02 per year, a number comparable to or greater than the frequency of offsite power failure. No measures to mitigate the water side corrosion of the backup cooling coil were indicated in section 3.2, whereas section 3.5.1.4 outlines sound procedures for maintaining the backup cooling coil and bringing it on stream in case of emergencies. It must be noted that the corrosion rate of type 304L stainless steel in nitric acid or nitrates is low, but pitting corrosion from the water side may occur readily either due to the presence of aggressive species such as chloride (Sridhar et al., 1987) or due to microorganisms growing in stagnant water in the backup cooling coil (Videla et al., 1994).

3.2 UNMITIGATED CONSEQUENCE OF BOILING IN THE Cs STORAGE VESSEL

BNFL, Inc. calculated the consequence of self boiling in the Cs storage vessel that could arise from a breach of the process vessel vent system (PVVS), which releases the radionuclides to the cell. Based on this calculation, the unmitigated consequence was 1.2 rem to workers, which classifies this accident as severity level (SL) 3. The frequency of occurrence of SL 3 events should be less than 1×10^{-3}. In section 3.5.2.3, however, the consequence from unmitigated release of Cs from the HLW mixing tank through a breached cooling coil was calculated to be 17,000 rem for the worker. Since the Cs storage vessel is about 10 times as voluminous as the HLW mixing tank and the cooling coils are expected to be sized proportionately, the unmitigated consequence from Cs release through a breached cooling coil would be expected to be much higher than that calculated for release through PVVS. This would place the hazard in the SL 1 category— requiring a much lower frequency of occurrence.

3.3 MITIGATED CONSEQUENCE

In selecting the control measures for the Cs storage vessel, BNFL, Inc. did not consider release through the cooling system. Therefore, control measures for mitigating the consequences, such as more robust cooling system or procedures for proper maintenance of the cooling water condition, were not considered. Further, mitigated consequences calculated by BNFL, Inc. for release through PVVS were based on the assumption that occupancy of the plant site under emergency conditions leading to power failure is unlikely to be high. This assumption may not be conservative, since one of the control measures (periodic addition of water to prevent boiling) requires the presence of operating personnel, and the control measures adopted for other systems such as the HLW mix tank (section 3.5.1.4) require the presence of personnel to bring the backup cooling system on stream properly.

3.4 INFLUENCE OF POWER LOSS ON CONTROL STRATEGY FOR LOSS OF COOLING TO THE Cs STORAGE VESSEL

In section 3.2.2.2, page 3.2-4, bullet one, the main report states that loss of power is the most likely cause of failure of both normal and emergency cooling water systems. Paragraph one, page 2, Safety Implementation Note (SIN) #99-00002,[1] indicates that backup power was not needed to meet the target unmitigated frequency estimate for loss of tank cooling for 3 days, and therefore, it does not have to be part of the control strategy (and it is not). One of the three elements of the selected control strategy (maintain vessel concentration to assure heat transfer) appears to rely on powered systems to deliver water to the vessel, operate an evaporator, or both. Power will also be necessary to run monitoring equipment and instrumentation. It is unclear how a control strategy for a loss of power dominated event can include SSCs that rely on power. The text of section 3.2.5.6 indicates that the frequency of unmitigated consequences includes credit for the operator adding water to the tank when all cooling is lost. It is not evident whether the systems for adding water would require the use of power and, if so, the analysis should consider how the unmitigated consequences would change if credit could not be taken for adding water to the tank.

Also, section 3.2.5.5 indicates that the consequences of failure to maintain concentration (i.e., add water) in the vessel are as described in section 3.4.2.3. The direct relevance of the feed line failure consequences (section 3.4.2.3) to a Cs tank boiling event is not readily apparent. Furthermore, the unmitigated consequence from the feed line failure is much higher (23 rem) than reported for the unmitigated loss of cooling consequence (1.2 rem) and is associated with SL 2 rather than SL 3, used for the loss of cooling control strategy assessment in section 3.2. Again, it is not evident from the description whether the means of maintaining concentration in the Cs vessel will require power to function or what the consequences are if loss of power causes the concentration management aspect of the control function to fail. Although the loss of power frequency is not high, providing backup power to the Cs storage cooling system appears a logical approach to mitigating this most likely cause of failure of the cooling systems, consistent with the principles of defense-in-depth and as low as reasonably achievable (ALARA).

3.5 REFERENCED RELIABILITY DATA

Page 2 of SIN #99-00002[2] refers to the five references on page 3 for information regarding reliability data. Key information and assumptions such as the bases for the reported failure probabilities, which should be contained in the reports, are referenced to a variety of reports. The description of the reliability assessment should be sufficiently complete so that the assessment could be verified by a reviewer. The present description does not indicate what information was used from the references and how the selected information was used in the reliability calculations. The same issue applies to SIN #W375–99–00021[3] cited in the section 3.5 cooling water contamination example.

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[1]Kolacskowski, A.M. 1999. *Accident Frequency/System Reliability Analysis for Loss of Cooling to the Cs Storage Vessel.* SIN #99-00002. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

[2]Ibid.

[3]Kolacskowski, A.M. 1999. *Accident Frequency/System Reliability Analysis for Cooling Water Contamination Example.* SIN #W375–99–00021. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

3.6 WATER RELATED SYSTEMS, SUBSYSTEMS, AND COMPONENTS

Section 3.2.3.3 lists one element of the control strategy for loss of coolant water to the Cs storage vessel as the addition of water to maintain the concentration below a specified level. The Cs storage vessel water makeup line is listed as an SSC that implements the control strategy, but there is no mention of the systems that provide the water supply to the line (e.g., source, delivery, or storage systems). It is not evident from the description how the failure of such systems would impact the availability of the needed control function.

3.7 ADDITION OF WATER/PASSIVE COOLING

Section 3.2.5.2.5 states that the vessel design and the maximum allowable concentration provide passive cooling; thus, loss of power will not result in self-boiling. Prior discussion in the report indicates that the control of the vessel concentration below the maximum allowable concentration requires periodic addition of water by a worker. It is unclear how the addition of water by a worker to maintain concentration in the vessel can be described as a passive system.

4 COMMENTS ON VOLUME II, SECTION 3.3, LOAD DROP OF A PRETREATMENT PUMP (OUT OF CELL)

4.1 REJECTION OF RESPIRATORS AS A CONTROL OPTION FOR LOAD DROP

Section 3.3, table 3.3-1, notes rejection of the use of respirators as a control option to protect operators in the event of a load drop of a pretreatment pump during transfer for maintenance. Given the high calculated consequence from a drop (46 rem) and that inhalation is 99.9 percent of the calculated unmitigated worker dose, use of respirators would appear a logical approach to providing defense-in-depth and maintaining exposures ALARA. The rationales for rejecting the use of respirators (i.e., operationally undesirable, does not prevent release, and depends on operator actions) as stated, are less convincing in light of the benefits to worker safety.

4.2 FILTER AT CORRIDOR VENTILATION AIR EXIT

The postulated pump drop accident in this section would result in a calculated unmitigated dose consequence from inhalation of 46 rem (SL 1) to the facility worker located in the corridor in the vicinity of the drop. Further, the corridor in which the accident is postulated to occur is serviced by the C2 ventilation system, which also services other normally occupied operation areas in the plant (British Nuclear Fuels Limited, Inc., 1998). It is not evident from the BNFL, Inc. description if a filtration system is provided at the air exit location for the ventilation air from the corridor. Such a system would minimize the impact from the spread of airborne contamination to the other normally occupied areas located downstream in the C2 ventilation system.

4.3 FLASK DESIGN FOR CORROSION RESISTANCE

The primary design requirement for the flask is that containment and shielding features must be maintained post drop (section 3.3.4.2.3). Further, to provide the design target decontamination factor of 10, the flask must withstand chemical attack from any of the acidic or alkaline wash liquors that may have been used on or with the pump. The description of the flask provided by BNFL, Inc. in section 3.3, however, does not adequately specify the construction materials (i.e., stainless steel, carbon steel, or titanium), the fabrication methods (e.g., welding type), the metallurgical conditions of the construction material (e.g., work hardened or stress relieved), nor the environmental conditions beyond the potential pH range of acidic to alkaline. Better definition of the design aspects of the flask and a more detailed description of the anticipated in-service environment (e.g., likelihood of the presence and concentration of halides or better definition of the pH range), are required to make an informed evaluation of the potential for failure of the flask and release of potentially radioactive effluent.

5 COMMENTS ON VOLUME II, SECTION 3.4, HIGH-LEVEL WASTE MELTER FEED LINE FAILURE

5.1 AIR DISPLACEMENT SLURRY PUMP INTERLOCK SYSTEM

The control strategy selected by BNFL, Inc. (section 3.4.3.2.3) includes a flow detection instrument that is interlocked to shut off the air displacement slurry (ADS) pump(s) in the event of a line break. The break is detected by the resultant loss of flow signal while pumping (figure 3.4-2). An ADS pump uses compressed air to propel a slug of slurry through the discharge piping. It therefore produces pulsating flow consisting of a series of slurry/air/no-flow segments in sequence in the melter feed line. The pulsed flow pattern generated by the ADS pump will contain several periods of no-flow of varying time durations, depending on the melter feed rate. In addition, no-flow in the the HLW feed line caused by a break in the pipe is likely to be mistaken for no-flow due to line plugging—a common occurrence in HLW feed lines. It is not evident from the description what type of flow sensor is being proposed by the BNFL, Inc. to distinguish between a no-flow condition caused by a break in the feed line and a no-flow time duration in the feed line during normal operations or during plugging. If the flow detection system is not capable of discriminating feed line failure, it would result in the failure of the BNFL, Inc. control strategy.

5.2 ATOMIZATION OF THE HIGH-LEVEL WASTE FEED

A break in the HLW feed line is likely to result in slugs of slurry atomizing in the cell atmosphere as they are being forced out of the pipe under around 50 psig pressure by the compressed air. It is not evident whether the BNFL, Inc. calculations take this into account.

6 COMMENTS ON VOLUME II, SECTION 3.5, COOLING WATER CONTAMINATION

6.1 MISSING KEY REFERENCE FOR CONSEQUENCE CALCULATION

In section 3.5, page 3.5-8, paragraph one, the referenced calculation (CALC–W375–HV–NS00002)[1] regarding unmitigated consequence calculation for cooling water contamination event, was not included in the referenced materials.

6.2 REJECTION OF GAMMA DETECTOR CONTROL OPTION

Section 3.5, page 3.5-10, lists controls for the cooling water contamination example. The list includes two control options: providing a gamma detector on the cooling line and adding an interlock on the cooling line. The engineering screening process removes the gamma detector from the control strategy but includes the interlock option. Because the interlock control option is triggered by a gamma detector, it appears that by selecting the interlock control option the gamma detector option is satisfied completely. Table 3.5-4 (page 3.5-16), however, shows no further consideration for the gamma detector control option. This could be misleading and perhaps requires further clarification. It is not clear whether the two options are sufficiently different to warrant consideration as separate options and, if so, whether the gamma detector control option is satisfied when the interlock option is selected.

6.3 COOLING SYSTEM DESIGN

In section 3.5.3.1, one of the control measures considered was the addition of a tertiary cooling loop that would be contained within the process cell and with which the secondary loop would exchange heat. The secondary loop would be located outside the cell. In the initial screening process, this option was ruled out (table 3.5.3). In sections 3.5.4.1.1 and 3.5.5.3, however, the small dose was assumed to result from the very small leaks in the "intermediate loop" of the cooling system. It is not evident from the description provided by the BNFL, Inc., whether intermediate loop refers to the secondary or tertiary coils.

Another control option is the use of an online radiation monitor in the secondary cooling water with interlocks to the pumps such that cooling water flow can be stopped if a leak is detected. The cooling water stoppage may raise the chances of boiling, as analyzed in section 3.2 with respect to the Cs concentrate tank. Because the HLW blending tank is one-tenth the size of the Cs storage tank, the heat transfer efficiency of the tank will be poorer. One of the control measures for the Cs storage tank was to increase its size.

6.4 CONTROL STRATEGY INVOLVING AREA RADIATION MONITORS

In section 3.5.3.2.3, BNFL, Inc. indicated that area radiation monitors will provide warning to the facility workers to evacuate an area in case of leaks from cooling pipe failure. To prevent self boiling, however, facility workers may be needed to bring the back-up cooling coil on stream using the controlled procedure outlined in section 3.5.4.1.3.

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[1]Smith, D.A. 1999. *Cooling Water Contamination.* CALC–W375HV–NS00002. Revision 0. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

7 COMMENTS ON VOLUME II, SECTION 3.6, SAMPLE CARRIER BREAKOUT

7.1 SELECTION OF RADIONUCLIDE INVENTORY FOR WORST-CASE SAMPLE

BNFL, Inc. used combined envelope B and D feed as the worst-case sample for dose rate calculations. A concentrated HLW feed containing washed HLW solids mixed with Cs-137 and Tc-99 solutions may provide higher dose rates than the combined envelope B and D feed. The dose rate could further increase if excess Cs-137 or Tc-99 from the storage tanks are inadvertently transferred to the HLW blending tank (V32004). BNFL, Inc. used the Best-Basis inventory to estimate a dose rate of 0.84 rem to the worker. It is not evident from the description provided whether this dose rate to the worker would significantly increase if the worst-case scenario, which includes an upper bound of the radionuclide inventory (i.e., 90th percentile) with higher concentrations of Cs-137 and Tc-99 in the tank, is used for estimation.

7.2 SELECTION OF PNEUMATIC TRANSFER PIPING

Section 3.6.1.1.2 indicates that BNFL, Inc. selected unplasticized polyvinyl chloride (uPVC) for the pneumatic transfer piping and that the selection process included an evaluation of alternative materials, including stainless steel. Under the heading, Improved Piping Material, concerning Operability, table 3.6-2 states, "System does not have experience with alternative pipe material." Similarly, concerning Maintainability, table 3.6-2 states, "System experience not available with alternative pipe materials." In the United States, however, stainless steel piping is routinely used at the West Valley Demonstration Project (WVDP) and the Savannah River Site (SRS) for such systems because stainless steels provide quicker decontamination and decommissioning, better integrity for lines exposed to external atmospheric conditions, and better toughness. Although it is understood that BNFL, Inc. has routinely used uPVC piping for pneumatic transfer piping with no major problems, the information cited above does not provide assurance that uPVC is the best material for the pneumatic transfer piping at the Hanford site.

7.3 SELECTION OF SAMPLE BOTTLES AND TRANSFER SYSTEM

Section 3.6.1.1.2 indicates that BNFL, Inc. intends to use robust high-density polyethylene (HDPE) bottles and a Sellafield-type transfer system (see section 7.2 above) for containing and transferring radioactive feed samples. In the United States, however, glass bottles are routinely used at the WVDP and the SRS as such sample bottles because they are more radiation resistant and chemically durable than HDPE. While BNFL, Inc. has apparently used HDPE for this purpose successfully at Sellafield, the decision to use HDPE sample bottles and a Sellafield-type transfer system rather than glass sample bottles and a United States-type transfer system at the Hanford site has not been adequately addressed.

8 COMMENTS ON VOLUME II, SECTION 3.7, LOW ACTIVITY WASTE PIPE BREAK

8.1 FAILURE TO DEENERGIZE THE TRANSFER PUMP

In section 3.7.1.2, BNFL, Inc. states that the leak detection system will be interlocked to automatically deenergize the transfer pump if a leak is detected. While BNFL, Inc. has not selected the final method for leak detection, a signal from the leak-detection system will be required to deenergize the transfer pump. In section 3.7.3.3, BNFL, Inc. has indicated that electrical power is not relied on to implement the control strategy as a performance requirement of the leak detection system to stop the transfer on loss of power, but in table 3.7-8 it is stated that shut down of the transfer pump on loss of electrical power to the leak detection system is a design safety feature. It is not evident from the description whether the components of the leak detection system and its electrical (or battery) power supply are important to safety or not.

8.2 STANDARDIZATION WITH EXISTING DESIGN SAFETY FEATURES IN THE HANFORD TANK FARM

Maintaining consistency between the proposed BNFL, Inc. DSFs and the existing DSFs implemented by other operators such as Fluor Daniel at the Hanford site tank farm is extremely important. For example, current practice at the Hanford site includes DSFs such as 6-in.-wide tape buried 1 ft below grade over pipe runs, and operator surveillance of transfer line. These controls have been excluded by the BNFL, Inc. Absence of these controls could result in unanticipated accidents due to human error especially if the workers from other Hanford areas, accustomed to the control strategy that includes 6-in.-wide tape buried 1 ft below grade, conduct the work. These workers may be looking for control strategies that do not exist in the BNFL, Inc. area and ignore the ones implemented by the BNFL, Inc. BNFL, Inc. has not identified possible conflicts between their control strategies and the existing DSFs at the Hanford tank farm.

9 COMMENTS ON VOLUME II, SECTION 3.8, RECEIPT VESSEL RUPTURE

9.1 CONCENTRATION OF Sr-90 IN HIGH-LEVEL WASTE RECEIPT TANK

In section 3.8.1.1 and table 3.8-2, BNFL, Inc. assumed that 90 percent of the HLW from Tank 241–AZ–101 will be transferred to the HLW receipt vessel. In table 3.8-3, however, 100 percent of the Sr-90 is assumed in the HLW receipt tank. It is not stated in the description why 90 percent Sr-90 that would have been transferred from Tank 241–AZ–101 is changed to 100 percent Sr-90 in the HLW receipt tank.

9.2 RADIONUCLIDE INVENTORY UNCERTAINTIES

In section 3.8.2.3, BNFL, Inc. estimated 60 percent of the total radioactivity from Tank 241–AZ–101 to be present in one of the HLW receipt vessels. To obtain the most conservative assessment of radionuclide concentration in the tank, BNFL, Inc. assumed the HLW receipt vessel is full to its overflow volume. The uncertainty estimates associated with Sr-90 and Am-241 inventories, which are the most dominant species contributing to dose, have been neglected, however. This could significantly increase the dose.

9.3 SOLUBLE RADIONUCLIDES

In this example, BNFL, Inc. assumed that 100 percent of the soluble radionuclides will be removed from the solids by ultrafiltration. Experimental data to show 100 percent removal of soluble radionuclides is not provided to support this basis.

9.4 INCONSISTENT USE OF DATA

In section 3.8.2.3, BNFL, Inc. estimated unmitigated consequences from the tank rupture. The calculations assume that the vessel is filled to its overflow level, which will allow 60 percent of the washed solids in one vessel. The dose to the various receptors was calculated by decaying the current radionuclide inventory to year 2006 as shown in table 2.0, CALC–W375PT–NS00001.[1] While these assumptions are reasonable, they are inconsistent with the data presented in tables 3.8-1 through 3.8-4, which assume transfer of 90 percent solids from Tank 241–AZ–101 and the radionuclide inventory estimates based on year 1994. BNFL, Inc. has not stated why these tables are internally inconsistent.

[1]Kummerer, M. 1999. *Assignment of Severity Levels for Receipt Tank Rupture.* CALC–W375PT–NS00001. Richland, WA: British Nuclear Fuels Limited, Inc. Unpublished report.

10 COMMENTS ON VOLUME II SECTION 3.9, ACTIVITY BACKFLOW FROM A PROCESS VESSEL INTO THE VESSEL WASH CABINET

No comments.

11 COMMENTS ON VOLUME II, SECTION 3.10, NITRIC ACID HANDLING

11.1 OVERFILL DETECTOR AND ALARM

The proposed BNFL, Inc. design has a high-level alarm that will sound when the nitric acid storage tank is approximately 80 percent full, thus prompting the operator to stop the acid transfer before the tank overfills (table 3.10-3). While this strategy will reduce the probability of tank overflow, it does not eliminate the possibility of overflow due to human error. If this procedural control was supplemented with automatic control (i.e., if the level sensor is interlocked to automatically shut off the transfer pump at approximately 90 percent full), it would effectively eliminate the possibility of tank overflow, thus greatly enhancing the margin of safety. The BNFL, Inc. overfill detector and alarm strategy, as stated, has a potential for human error.

11.2 REMOTE ACCESS EMERGENCY SHUTOFF

The remote access emergency shutoff feature, as described in table 3.10-3 and sections 3.10.3.1 and 3.10.3.3, is designed to enable the operator to remotely shut off the nitric acid transfer upon detection of a spray or spill, by shutting off the transfer pump. While the shutoff of the diaphragm pump will prevent backflow of acid from the storage tank toward the tanker truck, it will not impede flow of acid from the truck into the transfer pipe. Thus, if there is a leak in the transfer line, acid will continue to flow from the tanker truck through the stopped pump into the environment.[1] To stop the transfer, it will be necessary to shut off both the pump and the supply valve at the tanker truck. The BNFL, Inc. remote access emergency shutoff strategy does not include the capability for remote shutoff of the supply valve at the tanker truck.

11.3 TRANSFER LINE PURGE

In section 3.10.1.4.3, it is recommended the unloading line be purged after the nitric acid transfer has been completed. It is assumed compressed air will be used to purge the line, the hose to the tanker truck will be disconnected, and the fitting capped. This situation could result in an incident similar to one that occurred at WVDP where the transfer line was pressurized due to a slow leak in the purge air valve and the hose cap hit an operator during tanker truck hookup operations. Such a situation could be avoided by drilling a weep hole in the cap to release pressure. BNFL, Inc. did not specify this as a potential design safety consideration.

11.4 TANKER TRUCK UNLOADING PAD

The BNFL, Inc. design for the tanker truck unloading pad does not consider design features to isolate the truck tires from prolonged contact with acid spills. The 12 M nitric acid can react vigorously with organic combustible material such as the rubber used in the tires. BNFL, Inc. did not specify this as a potential design safety consideration.

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[1]Note: The statement in section 3.10.1.1.3 that states that a shutoff diaphragm pump acts as a stop valve preventing the flow of fluid through it, is inaccurate.

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12 REFERENCE

British Nuclear Fuels Limited, Inc. 1998. *Technical Report.* BNFL–5193–TR–01. Revision 0. Richland, WA: British Nuclear Fuels Limited, Inc.

Sridhar, N., J.B.C. Wu, S.M. Corey. 1987. The effect of acid mixtures on corrosion of Ni-base alloys. *Materials Performance* 26: 17–23.

Videla, H.R., F. Bianchi, M.M.S. Freitas, C.G. Canales, and J.F. Wilkes. 1994. Monitoring biocorrosion and biofilms in industrial waters: A practical approach. *Microbiologically Influenced Corrosion Testing.* J.R. Kearns and B.J. Little, eds. ASTM STP 1232. Philadelphia, PA: American Society for Testing and Materials: 128–137.