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DNF SAFETY BOARD

Department of Energy
National Nuclear Security Administration
Washington, DC 20585

July 1, 2002

The Honorable John T. Conway
Chairman
Defense Nuclear Facilities Safety Board
625 Indiana Avenue, NW
Suite 700
Washington, D.C. 20004-2901

Dear Mr. Chairman:

This letter is in response to your letter of April 23, 2002, concerning start up of the Los Alamos National Laboratory (LANL) Aqueous Recovery Line for Plutonium-238 (Pu-238) scrap. Your letter identifies concerns related to hazard identification and analysis, and the use of administrative controls. Specific examples of incomplete hazards analyses, and improper safety controls are cited in a Staff Issue Report, dated April 9, 2002. The Board suggested that further evaluation of the safety basis and safety-related controls for the Pu-238 aqueous recovery line is warranted.

Based upon a thorough review of the concerns raised in the Staff Issue Report, LANL and the National Nuclear Security Administration (NNSA) have concluded that: 1) the hazard identification and analysis for Pu-238 aqueous recovery meet applicable standards and guidelines; 2) the application of controls to mitigate risk are consistent with applicable requirements and guidelines; and 3) Technical Safety Requirements established for these activities adequately capture relevant safety controls, in accordance with applicable requirements and guidelines.

In a letter dated June 3, 2002 (Enclosure 1), LANL addresses the specific concerns identified in the Staff Issue Report. Specifically, LANL provides clarifying discussion and analyses related to deflagration, resin accidents, mechanical hazard, chemical hazard, reliance on administrative safety controls, and additional observations made by the Defense Nuclear Facilities Safety Board (DNFSB) staff. The Office of Los Alamos Site Operations (OLASO) provides additional discussion (Enclosure 2) related to other issues raised in the Staff Issue Report and in subsequent discussion with the DNFSB Site Representative.

The NNSA has determined that the Process Hazard Analysis for the Pu-238 scrap line and the safety controls identified in the authorization basis documentation are adequate to support safe operations. However, the NNSA and LANL are committed to continuous improvement in the development of authorization basis documents and would welcome additional discussion with the DNFSB to identify opportunities for improvement.

Sincerely,

Everet H. Beckner
Deputy Administrator
for Defense Programs

2 Enclosures

cc w/enclosures:
M. Whitaker, S.3-1

Enclosure 1

02-1368

Los Alamos
NATIONAL LABORATORY

Office of the Director

June 3, 2002

Mr. Dennis Martinez
Area Manager
Office of Los Alamos Site Operations
528 35th Street
Los Alamos, NM 87544

Subject: Laboratory Response to DNFSB Letter Regarding Pu-238 Scrap Recovery Line

Reference: *Letter from J. T. Conway, Chairman Defense Nuclear Facilities Safety Board to General J. A. Gordon, Under Secretary of Nuclear Security and Administrator of the National Nuclear Security Administration, April 23, 2002*

Dear Mr. Martinez:

This letter responds to the Defense Nuclear Facilities Safety Board (DNFSB) letter of April 23, 2002. The referenced letter requests resolution of deficiencies that were identified in its Staff Issue Report of April 9, 2002, relative to:

1. Hazard identification and analysis for this activity,
2. Use of administrative controls where engineered controls are available and use of mitigative controls where preventive controls are available, and
3. Specification of Technical Safety Requirements (TSRs) that capture all of the relevant safety controls identified in the approved authorization basis, consistent with DOE directives.

The information in this response was discussed and coordinated with the National Nuclear Security Administration (NNSA), the DOE Office of Los Alamos Site Operations (OLASO), and the Los Alamos National Laboratory. For reference, pertinent quotes are provided from DOE-STD-3009, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Report*. The response identifies and addresses the DNFSB specific areas of concern including deflagration, resin accidents, a mechanical hazard, a chemical hazard, implementation of safety controls, and additional observations.

Sincerely,

John C. Browne
Director

P. O. Box 1663, MS A100, Los Alamos, NM 87545
[illegible]
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National Nuclear Security Administration of the U.S. Department of Energy

Attachment: LANL Response

Cy: General John Gordon, Undersecretary, Nuclear Security Administrator, NNSA
Dr. Everett Beckner, Deputy Administrator Director, Defense Programs NNSA
R. Mah, AD-WEM, A107
T. George, NMT, E500
IM-5, A150
DIR-02-155 file

LANL Response to DNFSB Letter
Regarding Pu-238 Scrap Recovery Line
June 3, 2001

Introduction: The information in this response was discussed and coordinated with National Nuclear Security Administration (NNSA) U. S. Department of Energy Los Alamos Site Operations (OLASO) and represents consideration by Los Alamos National Laboratory (LANL) and the NNSA. The Defense Nuclear Facilities Safety Board (DNFSB) letter of April 23, 2002, to J. A. Gordon[1] requests resolution of deficiencies identified in its Staff Issue Report of April 9, 2002,[2] relative to:

1. Hazard identification and analysis for this activity.
2. Use of administrative controls where engineered controls are available and use of mitigative controls where preventive controls are available.
3. Specification of Technical Safety Requirements (TSRs) that capture all the relevant safety controls identified in the approved authorization basis, consistent with DOE directives.

Background: Before addressing the discussion and conclusions contained the DNSFB Staff Issue Report,[2] it would be prudent to present pertinent quotes from DOE-STD-3009[3] for reference.

<u>DOE-STD-3009, pg xxiii, Definitions</u>
Safety-significant structures, systems, and components (safety-significant SSCs), "Structures, systems, and components not designated as safety-class SSCs but whose preventive or mitigative function is a major contributor to defense in depth (i.e., prevention of uncontrolled material releases) and/or worker safety as determined from hazard analysis. As a general rule of thumb, the safety-significant SSC designations based on worker safety are limited to those systems, structures, or components whose failure is estimated to result in an acute worker fatality or serious injuries to workers. Serious injuries, as used in this definition, refer to medical treatment for immediately life-threatening or permanently disabling injuries (e.g., loss of eye, loss of limb) from other than standard industrial hazards. It specifically excludes potential latent effects (e.g., potential carcinogenic effects of radiological exposure or uptake). The general rule of thumb cited above is not an Evaluation Guideline. It is a lower threshold of concern for which safety-significant SSC designation may be warranted, not quantitative criteria. Estimates of worker consequences for the purpose of safety-significant SSC designation are not intended to require detailed analytical modeling. Considerations should be based on engineering judgment of possible effects and the potential added value of safety-significant SSC designation."

DOE-STD-3009, Page 10, TSR and SSC Commitments
"TSRs assigned for defense in depth or safety-significant SSCs (i.e., not related to meeting Evaluation Guidelines) do not have SLs and are not required to use operational limits (i.e., LCSs, LCOs). They should, however, receive coverage in the administrative control section of TSRs as a minimum. Judgment should be used to determine what controls warrant use of operational limits. When TSR administrative controls are used for purposes other than generic coverage of safety management programs, descriptions should be sufficiently detailed that a basic understanding is provided of what is controlled and why. Beyond safety-significant SSCs designated for worker safety and their associated TSR coverage, additional worker safety issues should be covered in TSRs only by administrative controls on overall safety management programs… safety-significant structures, systems, and components. This category of SSCs is provided to ensure that important SSCs will be given adequate attention in the SAR and facility operations programs. Safety-significant SSCs are those of particular importance to defense in depth or worker safety as determined in hazard analysis. Control of such SSCs does not require meeting the level of stringency associated with safety-class SSCs."

DOE-STD-3009, Page 67, System Evaluation
"Safety-significant SSCs, are not required to consider performance criteria traditionally associated with safety-class SSCs or traditional nuclear standards in general. Performance criteria for a safety-significant SSC should be representative of the general rigor associated with non-nuclear power reactor industrial and OSHA practices. Performance criteria for safety-significant SSCs are developed by SAR preparers using engineering judgment based on the expected functions for which it was designated a safety-significant SSC and its overall importance to safety. Evaluate the capabilities of the SSC to meet performance criteria. The evaluation should be as simple as possible, and rely on engineering judgment, calculations, or performance tests as opposed to formal design reconstitution. For example, the hydrogen detector could be fed a test gas composition that would exceed its interlock trip point. Such a test would typically bound the needed equipment performance, as response time is not a highly sensitive parameter."

Discussion: The DNFSB *Staff Issue Report*[2] identified inadequacies in the safety basis. Quoting from this report:

> Significant accident scenarios were apparently not adequately analyzed by LANL and DOE; the result was incomplete identification of controls in the safety basis. Contributing to this problem was the fact that DOE's approval memorandum made major changes to the safety basis without complete analysis and without requiring LANL to correct and update its PrHA. The deficiencies in the safety basis proposed by LANL might have been resolved more effectively had DOE held LANL responsible for reanalyzing the potential accident scenarios and developing a complete set of functionally classified safety controls.

Examples presented in the report included incompletely evaluated hazards, accident scenarios, and controls. Each example identified in the report is addressed in the following sections.

1. ***Deflagration*–Flammable gas generation by thermal and radiolytic decomposition received only a superficial evaluation, but it appears to be a major hazard warranting designated safety controls. For example, it may be appropriate to designate as a safety system the argon sparge that purges flammable gases from the dissolver system.**

In its original review of the Pu-238 Scrap Recovery USQ,[4] the NNSA commented on the flammable gas generation potential for this process. The following is a quote from the NNSA comments and the proposed LANL resolutions. NNSA/DP-45[5] Comment #3 on Page 2 of the attachment states:

> COMMENT. p. 66 of 91. Provide for a more detailed evaluation of the potential for H_2 generation and fire/deflagration. If only an insignificant amount of H_2 can be generated, indicate this by reference to the process contents and reactions.
>
> RESOLUTION. All scrap recovery processes were reviewed for potential of hydrogen generation and fire/deflagration. There are no processes that can generate hydrogen gas [i.e., flammable concentrations of hydrogen gas]. In Table 3, *Hazard Identification*, on page 66, the entry *Flammability and Fires* will be expanded to reflect this fact and the following under the heading Presence of Fuel: gloves and cheesecloth. Cheesecloth is stored in slip-top cans when not in use and thus removed from potential ignition sources.

This comment and others from the NNSA original review indicate that flammable gas generation in the scrap recovery process received more than a superficial evaluation. This USQ also received an independent review by NNSA members in OLASO and separately from DP-45; all comments were addressed and closed. The following explanation discusses the technical potential for hydrogen gas generation.

Hydrogen gas generation by alpha radiolysis in nitric acid solutions has been studied at Pacific Northwest Laboratory,[6] Savannah River,[7,8] Rocky Flats,[9] and in the USSR[10] and Japan.[11] Results are reported as G values, the number of molecules of gas generated per 100 eV.

In general:

- $G(H_2)$ decreases as $[HNO_3]$ increases, as H^+ atoms are ionically paired with NO_3^-.
- The amount of gas evolved is generally less than the amount of H^+ generated, and is increased by agitation.
- The volume of gas evolved varies with the depth of solution, with narrower and deeper vessels encouraging H^+ reaction with the solution, resulting in less gas evolution.
- It appears that H_2 generation is less for Pu nitrate solutions compared to Cm and Po, at the same radiation doses, but it is not possible to decouple this phenomenon from the agitation effect.

In the *Technical Basis for Maintaining Safe Hydrogen Levels in HB-Line Process Vessels,*[8] Dickson summarizes calculations on hydrogen gas buildup in the HB-Line. Dickson uses the ^{244}Cm results of Bibler,[7] whose test results were from agitated solutions, thus minimizing the amount of hydrogen dissolved in the liquid and maximizing the amount of gas evolved. Dickson calculated the time to reach 4% H_2, under static conditions at the maximum operating conditions in their OSRs, at 45 minutes for the dissolver. Savannah River maintained the H_2 level below 1% by providing an air purge to the vessel, with an LCO requiring in the event of loss of purge gas, either restoration of the purge or "a comparable action to a tank so that under maximum operating conditions the hydrogen levels will remain below 4%." Note that 4% in air is the lower flammability limit (LFL) for H_2.

Smith[12] used the Marquardt-Levenberg algorithm to fit the Bibler and other $G(H_2)$ data from the range 0 to 10.1M HNO_3^-. The following equation was used by Los Alamos to predict the bounding H_2 case:

$$G(H2) = 1.217/(1+1.702[NO_3^-])$$

We reported these results in memo NMT-9/02-036 dated April 8, 2002.[13] The calculations were performed:

- using isotopic analyses of several representative feed lots;
- assuming 300-gram oxide feed batches;
- assuming maximum H_2 gas generation and evolution; and
- taking no credit for dilution effects due to the argon flush in the dissolution vessel nor glovebox ventilation, nor for the higher flammability limit for hydrogen in argon.

The calculations show that the maximum H_2 generated in our 10-hour dissolution run is approximately 0.4 liters, and that after 30 days, approximately 30 liters is the maximum amount generated. The dissolution vessel is directly open to the glovebox, and the volume of the glovebox is approximately 4600 liters. Thus, even after 30 days and in the most conservative case (including complete evolution, as gas, all hydrogen generated, and no purging), the maximum H_2 concentration in the glovebox is less than one-fifth of the LFL.

The question of ability to vent through the scrubber to the glovebox remains. The dissolution vessel is vented to the glovebox through a water scrubber with an internal diameter of 6 inches and a height of 12 inches. During normal operations, the scrubber will typically be filled to a maximum height of 8 inches. However, we assume a 12-inch depth, which results in a head pressure of 0.43 psi or 0.0292 atm. The minimum headspace volume, determined for dissolving 300 grams of oxide in 3 liters of HNO_3 in the 4.2-liter vessel, is 1.2 liters. The ambient pressure above the solution is 0.77 atm (glovebox negativity having a negligible effect on ambient pressure for these calculations), and the maximum H_2 concentration in the headspace is 0.0292 atm/0.77 atm, or 3.8%. While this approaches the LFL of 4%, numerous conservatisms have been used in the calculations, and the volume is only 1.2 liters.

Thus, there is no credible scenario for build up of hydrogen gas at unsafe levels, and this was correctly treated in the hazard and accident analyses as a nonpotential. As shown by these calculations, the potential for hydrogen deflagration even after 30 days is impossible because the lower flammable limit for hydrogen in air (approximately 4%) is never reached. However, the process hazard analysis (PrHA)[14] conservatively identified the argon sparge as a defense-in-depth control. Based on the evidence supplied to the NNSA and their review of this process, the NNSA concurred with the categorization of the argon sparge as a defense-in-depth control, not a safety-class or safety-significant control.

2. ***Resin Accidents*–There are two principle examples of the incomplete analysis of scenarios related to anion exchange column accidents:**

    A. **As a condition of approval, DOE required that prevention of resin dry out be treated as a safety-class control. However, LANL has not identified safety-class controls that would be effective in the event of a facility power failure and subsequent building evacuation. The anion exchange columns have an autoelution system, powered by an uninterruptible power supply, that might serve this function. It may be appropriate to evaluate this as a potential designated safety system and thereby ensure its reliability.**

    B. **DOE designated the mesh screens around the anion exchange columns as safety-significant, presumably as mitigation against missile fragments that could breach the glovebox in the event of column over pressurization. However, preventive safety-significant controls such as relief valves or rupture discs did not appear to have been considered. The columns are equipped with relief valves, but the relief valves were not designated as safety systems and therefore may not be tested and maintained with the appropriate rigor."**

The Reillex-HPQ resin was selected for use because of its demonstrated superiority to other resins in the areas of chemical stability, resistance to radiolytic degradation, and thermal stability. Reillex-HPQ resin was designated as a safety-class control because it serves as a preventative feature in preventing overpressurizations that have occurred in past ion exchange column explosions.

With regard to the DNFSB Staff Issue Report[2] concern relative to the resin dryout issue during a facility power failure and subsequent building evacuation, the DOE approval memo[5] referenced a report entitled *The Effects of Ionizing Radiation on Reillex™ HPQ, a New Macroporous polyvinylpyridine resin, and on Four Conventional Polystyrene Anion Exchange Resins*, LA-11912, November 1990, by S. Frederic Marsh.[15] To assist in clarifying this issue, an excerpt from the report is reproduced here.

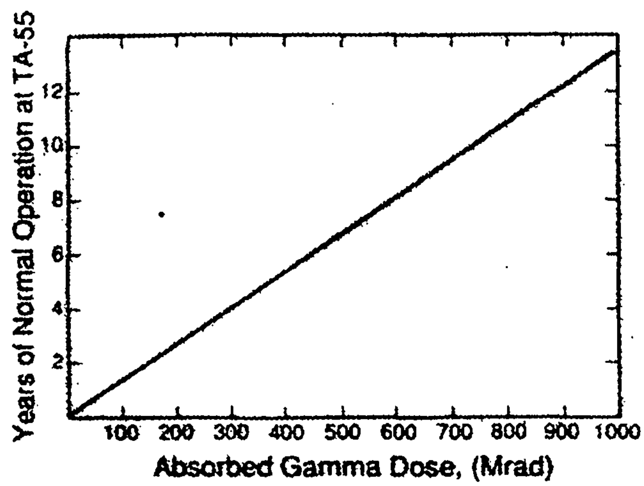

Fig. 10. Correspondence between absorbed gamma dose and years of normal operations at TA-55.

As shown in Fig. 10, an absorbed dose of 370 megarads corresponds to about 5 years of normal plutonium-processing operations at the Los Alamos Plutonium Facility (TA-55), which would indicate that the threshold for the observed thermal instability of Reillex™ HPQ resin could occur only after 5 years of routine operation at TA-55. (The Los Alamos Plutonium Facility, which operates a single shift five days per week and stores ion exchange columns in 1 M nitric acid between shifts, should not be used to estimate resin lifetime elsewhere.) Because resin damage from alpha particles is lower by at least a factor of two (see the following Alpha-Particle Irradiations section), however, the threshold for the cited thermal instability of Reillex™ HPQ resin actually corresponds to more than ten years of normal operations in the Los Alamos Plutonium Facility.

The combination of two factors—the improbability of resin becoming dried and heated during production operations and the required induction period of greater than 10 years at Los Alamos—virtually eliminates any possibility that thermal instability could occur in a production environment. Nevertheless, we recommend that Reillex™ HPQ resin be replaced after not more than 5 years of service, or 700 megarads of alpha-particle irradiation, or 400 megarads of absorbed gamma dose.

To summarize, use of the Reillex™ resin could become a problem if:

1. The age of the resin was greater than 10 years without being changed out due to poor TSR compliance year after year until 10 years was exceeded.
2. The resin became dried out (either due to a thermal environment like a fire or to poor TSR compliance).
3. The resin became heated (due to a thermal insult).

All these conditions must be simultaneously met for the resin to become thermally unstable. In the DNFSB Staff Issue Report[2] scenario:

- the resin is allowed to age for more than 10 years without ever being changed out,
- there is a facility evacuation,
- the resin dries out (or is already dried out),
- the resin becomes thermally unstable, and then simultaneously suffers thermal insult.

As identified by Marsh,[15] this scenario, or any scenario similar to it, is virtually eliminated by the need for all three scenarios listed above to occur simultaneously. The resin will not dry out during short-term offsite power outages, and TSR-level administrative controls will prevent the resin from being allowed to dry out during potential long-term shutdowns. These were contributing causes in past accidents.

The ion exchange column is equipped with an auto-elution system that is activated in the event that level sensors in the column detect that

- the solution level over the resin reaches a minimum value,
- the temperature of the ion exchange column reaches 55 °C;
- the pressure in the ion exchange column reaches 16 psi, or
- an auto-elute button is pressed.

During the auto elution, 0.45M HNO_3 is pumped through the column and the eluate is directed to a storage tank. In the event of a facility power outage, an uninterruptible power supply provides temporary power. The work instruction for ion exchange lists several pre-operational checks, including a visual check of the resin condition and operability of the auto-elution system.

The controls placed on prevention of dryout include implementing administrative controls, changing out the resin at a maximum lifetime of 5 years, and minimizing glovebox combustible loading. These controls and the fact that any scenario presenting the opportunity for thermal instability is virtually eliminated by the need for all three scenarios listed above to occur, leads to the conclusion that the auto-elution system would be extreme defense-in-depth and does not need to be considered safety-significant.

The mesh screens around the anion exchange columns were designated safety-significant for worker safety using qualitative engineering judgment according to DOE-STD-3009 (refer to the DOE-STD-3009 excerpts presented above). Both DOE-STD-3009 and DOE Order 420.1[16] state that selection of controls is judgment-based and depends on many factors, such as effectiveness, a general preference of preventive over mitigative and passive over active, relative reliability, and cost considerations. The mesh screens were designated safety-significant because they are a passive system that requires no maintenance, testing, or surveillance after installation, which serves to minimize cost and provides significant reliability and defense-in-depth. The mesh screens would serve to protect workers against flying fragments should personnel error occur and the columns become pressurized, somehow resulting in rupture of the columns. In this respect, the mesh screens provide a passive mitigative control for potential overpressure scenarios that, when combined with the preventive controls for overpressure, provide a comprehensive set of safety controls for scrap recovery aqueous operations.

Pressure relief valves (PRVs) and rupture discs are used to relieve pressure after the pressure transient has started, which makes this feature mitigative rather than preventive. Further, as described on page 75 in the PrHA,[14] during ion exchange processing, the columns are open at the top to prevent the buildup of excessive pressures. In fact, with the column open to the glovebox atmosphere, the PRV could not even function in this case. Therefore, the relief valves mentioned do not serve any purpose during ion exchange processing except to relieve pressures in the event of personnel error that results in the failure to leave the columns open during processing (a defense-in-depth function at best). The PrHA conservatively identifies the relief valves function as defense-in-depth controls that are mitigative in nature. Based on the evidence supplied to NNSA and their review of this process, NNSA concurred with the categorization of the relief valves as a defense-in-depth control, not a safety-class of safety-significant control.

Pressure relief devices are tracked on an NMT-9 database, and their use and replacement are governed by a controlled procedure. The procedure calls stipulates periodic replacement of representative valves, with post-replacement checks of operability. If deficiencies are found in any removed valve, the others will be removed, replaced, and checked for operability. DOE-STD-3009 states,

> By virtue of application of the graded approach, the majority of the engineered features in a facility will not be identified in the categories of safety-class or safety-significant SSCs even though they may perform some safety functions. However, such controls noted as a barrier or preventive or mitigative feature in the hazard and accident analyses must not be ignored in managing operations.

The tracking and periodic replacement and post-replacement testing of relief valves by NMT-9 complies with DOE-STD-3009 guidance and is an adequate control for safety.

3. ***Mechanical Hazard*–DOE designated the comminution ball mill jars as safety-significant, and the physical restraint systems (e.g., locking tabs, protective cover) that prevent ejection of the jars as defense-in-depth. Thus there is greater control on mitigation (the jars) than prevention (the restraint system). It is not clear that mitigative, rather than preventive, controls are preferred in this case. Furthermore, it does not appear that LANL evaluated whether the jars, if ejected, would adequately contain the plutonium oxide powder. It may be appropriate to investigate the restraint mechanisms as possible safety-related, preventive controls.**

The ball mill jar is secured in the cradle by tightening the top plate of the cradle using a setscrew. The setscrew is secured by a locking tab to prevent loosening. The locking tab is threaded (as a nut), with a tab or toggle protruding from the side. The locking tab is rotated around the screw threads until it is secured against the upper clamp of the cradle. The edges of the circular top and bottom plates of the cradle are formed inward to prevent a mill jar from sliding out of the cradle. In addition, the surface of these plates has a layer of hard but slightly compressible pads that provide increased friction, which also prevents the jar from sliding within the cradle.

The hazard associated with ejection of the ball mill jar is not from the flying jar, as will be shown later in this explanation, but rather from the release of the Pu-238 oxide contained within and posing an inhalation hazard to the workers. As is demonstrated below, the ball mill jars do not travel with sufficient speed/energy to breach the glovebox confinement system. The designation of the ball mill jars as a safety-significant SSC provides the second layer of protection for the workers by keeping the oxide contained within. Both these SSCs act as barriers and serve to prevent the release of the Pu-238 oxides. Therefore, the ball mill jar restraint system (a system relying on human intervention to work) appropriately provides a third layer of protection and its designation as defense-in-depth is in accordance with the precepts of DOE Order 420.1 and DOE-STD-3009, which emphasize the use of engineered controls over administrative controls.

The comminution process batch size is 300 grams of $^{238}PuO_2$. The process description in the PrHA[14] for comminution states that up to 50 grams of ^{238}Pu oxide are transferred into each ball mill jar. Note that, in practice, only 20 grams are loaded in the jar during processing. Two ball mill jars are loaded into the mill for each run. The unmitigated accident scenario for the ball mill considers the following:

- One batch of 300 grams of oxide has been completed and stored in the comminution glovebox, awaiting transfer to the dissolution glovebox.
- Another ball mill run is being completed.
- Each of the two ball mill jars contains a maximum of 50 grams of oxide, for a total of 100 grams of $^{238}PuO_2$.
- One or both jars are not secured in their respective cradles, or the locking tab is not engaged and the setscrew loosens.
- The cover of the ball mill is left open and the power interlock to the cover fails.
- The first mill jar is ejected and penetrates a glovebox window.
- The ball mill is extremely unbalanced, the cradle around the second mill jar loosens and, the second jar is ejected.
- The second mill jar impacts the powdered oxide container, the container lid comes off, and the oxide is released.
- It is assumed that the contents of both mill jars are also ejected.
- In addition, another 100 grams of oxide is assumed to be available in the glovebox for dispersal.
- The total dispersible material-at-risk for this accident scenario is 500 grams.

The Spex 8000D ball mill operates with two nearly identical ball mill jars. Each jar is secured in a separate cradle. Each jar, gaskets, and the two balls have a mass of 700 grams. The maximum mass of oxide loaded into a jar is 50 grams. Thus, the total mass of the jar and oxide is 750 grams.

The jar moves in a figure-8 motion. The total travel distance for one cycle is 4 inches. The jar moves at 1100 cycles per minute or 18.3 cps. The average velocity of the jar during each cycle is 73.3 inches per second or 1.86 m/s. It is assumed that the jar is ejected at the average velocity. The kinetic energy of the ejected jar is:

$$E = \frac{1}{2} mv^2 = 1.30 \text{ N-m} = 0.96 \text{ ft-lb}$$

In her PhD thesis, Wang[17] developed a 3-D mathematical model for the SPEX-8000 mixer/mill. Her model predicts a maximum ball velocity in this scenario of approximately 8 m/s. Assuming conservatively that a jar and its contents would be ejected at this speed, the maximum energy is calculated to be 24 N-m or 18 ft-lb.

Glovebox windows are made of Category II laminated glass that meets the impact requirements of the Consumer Product Safety Commission (CPSC). Category II glass windows meet the 400 ft-lb impact requirements of 16 CFR 1201. The test specimen for Category II glass is 34 inches by 76 inches. A typical window on the top of a glovebox is made of laminated safety glass with dimensions of 7-5/8 inches by 22-3/4 inches. The glovebox windows experience less deflection and tensile stress than the larger impact test specimen required by the CPSC.

While the impact from an edge of the metal ball mill jar would distribute the energy over a smaller area than that of the deformable surface of the test specimen impactor, the impact energy of an ejected ball mill jar is still less than the effective energy delivered to the test specimen. And, even if the glass components of the laminated glass window were to crack, the window would retain its physical integrity due to the laminating layer.

The metal ball mill jars can withstand significantly greater impact energies. For example, the Charpy impact energy for stainless steel is 148 ft-lb or even for aluminum it is 51 ft-lb, which is greater than the maximum ejection energy of a ball mill jar (18 ft-lb).

In addition, several safety features have been identified on the ball mill. These include the work instruction for operation of the ball mill, the ball mill jar, the crank (setscrew) that secures the ball mill jar in the clamps, the locking tab, and power interlock on the mill lid. Operator error could prevent the tightening of the setscrew and securing the locking tab. An interlock switch could also fail. The robust design of the manufacturer-supplied ball mill jar provides the most consistent safety control and was designated as safety-significant. The work instruction, setscrew, and the locking tab also are identified as defense-in-depth.

4. ***Chemical Hazard*–The laboratory intends to use hydroxylamine nitrate (HAN) as a reducing agent prior to oxalate precipitation. Under certain conditions, HAN has been known to undergo autocatalytic decomposition in the presence of nitric acid. In its approval memorandum, DOE pointed out that recommendations for safe handling of HAN were provided in the February 1998 *Technical Report on Hydroxylamine Nitrate, DOE-EH-0555*. The staff believes that HAN can be used safely in the scrap recovery process, provided that the recommendations in the technical report (particularly those regarding temperature, concentration, and storage criteria) are rigorously implemented. The laboratory has yet to demonstrate how it will implement the DOE recommendation; it may be appropriate to develop safety-significant controls to prevent accident scenarios involving HAN.**

Demonstration of *how* controls are implemented occurs in the procedures that are developed before the readiness assessment (RA) occurs. At this point in the life of this project, those procedures have not yet been developed. However, verification of *how* controls are implemented will be addressed in Laboratory's RA and the NNSA (RA) for scrap recovery before the process becomes operational. As demonstrated below, the low weight percent of HAN usage in scrap recovery and the low molar concentration of HNO_3 are well below the levels of concern for the occurrence of autocatalytic decomposition.

The ^{238}Pu aqueous scrap recovery line will use 24% w/v (2.8M) as received from the manufacturer. Unlike the ^{239}Pu aqueous process, HAN is not used to elute the ion exchange column, only during oxalate precipitation. Before HAN is added, a free acid titration is performed to calculate the amount of dilute nitric acid that is necessary to achieve a final concentration of HNO_3 in the 1 to 2 M range. The dilute nitric acid is prepared in the precipitation vessel, the solution from dissolution or ion exchange is added to the vessel, and the final pH is confirmed before proceeding. Holding agents followed by HAN, to reduce all the Pu in solution to the Pu(III) valence state, are added. The ratio of HAN to Pu is 6:1.

Report DOE/EH-0555[18] summarizes experiments that were conducted at Savannah River Site and Hanford to determine the autocatalytic (unstable) and unreactive (stable) zones for HAN in HNO_3, as a function of [HAN], [HNO_3], [Fe], and temperature. The Instability Index, I, is:

$$I = (1+[HNO_3])^{(1+\log[HAN]/[HNO_3])} + (1+[HNO_3])^{(1+\log(1+100[Fe]))}$$

A graph of decomposition temperature versus Instability Index is plotted in Figure 4 of reference 18.

We have used this algorithm to calculate I for several representative runs. The temperature measured during the process, on a 100-gram oxide batch, varied between 40 °C and 45 °C. The concentration of Fe was calculated assuming 1500 ppm in the feed, the highest value measured on feed processed through our bench-scale line. Note that I is relatively insensitive to [Fe], e.g., a factor of 100x increases I by approximately 2 units.

Values of I resulting from these calculations are between 4 and 5, well within the unreactive zone.

5. *Implementation of Safety Controls*–**The staff review identified concerns regarding the effectiveness with which the proposed administrative controls would be implemented. Based on discussions with LANL personnel, LANL plans to start up the scrap recovery process before incorporating many of the safety-related administrative controls—including controls that implement functions designated as safety-class by DOE—into Technical Safety Requirements (TSR). LANL intends to make such changes to the plutonium facility TSR in its 2003 annual implementation and surveillance requirements before startup of the process.**

The process hazards analysis (PrHA) *Process Hazard Analysis Aqueous Recovery of Plutonium-238 Scrap*[14] was submitted to DOE for review of the document and approval of the associated processes. That review and approval was documented in the December 1, 2000, memo *DOE Approval of the Process Hazard Analysis for the Aqueous Recovery of ^{238}Pu Scrap Oxide*, DOE memorandum SABT:3TW-029.[5] Technical Safety Requirement (TSR)-level programmatic controls were identified, safety-class controls were identified, safety-significant controls were identified, and several additional requirements were identified.

In accordance with the TA-55 Authorization Agreement, documented in Los Alamos National Laboratory memorandum dated December 22, 1999, from LANL Director John Browne to DOE Area Manager David Gurule',[19] the basis for authorization for TA-55 includes all facility modification and new work activities reviewed in accordance with the TA-55 USQ process defined in DOE Order 5480.21.[20] The Aqueous Recovery of Pu-238 is an operational change that followed and was approved according to the TA-55 USQ process. Aqueous Recovery of Pu-238 as well as its requirements defined by the DOE is part of the current TA-55 authorization basis.

The TSR-level controls identified by DOE as well as the consequence calculations will be incorporated into the SAR/TSR in the 2003 update, but they will also exist as part of the current

TA-55 authorization basis. They will be fully implemented and go through a readiness review to verify implementation before operations are fully authorized. Neither DOE-Order 5480.21 nor 10 CFR 830 Subpart B require revision to TSRs as the result of an approved USQ. It is prudent, and required, to revise TSRs if a new LCO is created, or if, for example, an LCO or a surveillance is modified as the result of the USQ approval, such as was done with the past approval of the TA-55 radiography USQ. However, this is not the case here and this USQ approval could be accomplished safely without imminent TSR modification. As stated earlier, approved USQs and completed USQDs become a part of the authorization basis once completed and approved as appropriate.

6. ***Additional Observations*–The staff observed that transfers of solutions will be accomplished using Tygon tubing temporarily routed through connecting doors between the gloveboxes. This practice appears to be questionable for two reasons. First, DOE specifically required LANL to minimize opening these doors to protect assumptions regarding material-at-risk (MAR). Although each solution transfer is expected to be of short duration, they still will violate the MAR assumptions. More importantly, performing transfers using temporary hoses attached with quick-disconnect fittings is not consistent with the production nature of this process (i.e., long-term, nearly continuous operation). This approach provides operational flexibility, but it may result in excessive contamination in the gloveboxes due to inadvertent drips and spills of solutions. LANL may be better served to investigate a solution transfer method that does not require repeated connection and disconnection of transfer lines.**

The introduction to the PrHA[14] states that doors between gloveboxes are normally closed and sealed. The DOE statement does allow transfers of material, equipment, and components. The doors are opened, connections are made, the transfer is completed and the Tygon tubing is moved into a single glovebox, and the doors are closed. Otherwise, solid precipitates from the oxalate or hydroxide precipitation processes could not be moved through the glovebox line for calcination. Solution transfers are also of short duration. The act of opening the doors between gloveboxes and moving items through the spool does not constitute a violation of MAR assumptions.

Regarding the production nature of this process, as described in the PrHA,[14] in the presentations to the DNFSB staff, and as evidences during the tour of the glovebox line, the aqueous recovery of ^{238}Pu scrap comprises several separate unit operations that are performed in a batch sequence. There is no continuous flow of material through the gloveboxes.

Use of Tygon tubing for solution transfers is an established technique. Solutions are moved between columns or tanks using a wet vacuum transfer process. The vacuum minimizes the amount of solution remaining in the Tygon tubing. Drips (~ milliliter) or small quantities of solution are wiped up. If spills occur, Tygon tubing attached to the wet-vacuum system is used to remove the solution and the glovebox provides contamination control.

The DNFSB observations imply that any leakage constitutes *excessive* contamination. Solution gloveboxes typically have lower contamination levels than gloveboxes that house powder operations. LANL has provided redundancy for most processes and components in each unit operation. The ability to work around any one component is an asset, and the operational flexibility provided by the Tygon tubing transfer should not be trivialized. The use of stainless steel tubing and valving necessary to provide comparable capabilities would introduce additional volume and complexity. More importantly, the use of Tygon tubing ensures that any leaks are contained within the glovebox boundary.

The scrap recovery operation is expected to process a maximum of 5000 grams of material per year in 300-gram batches (approximately 17 batches per year). Solution transfers occur between the filtrate storage tank in GB-1215 and the precipitation column located in GB-1217. The filtrate storage tank has a capacity of 5000 milliliters. The expected transfer flow rate is 10 ml/sec for a total transfer time of approximately 8.5 minutes per batch. Multiplying the number of batches per year (17) times the transfer time per batch equates to a total time of approximately 2.5 hours per year that the doors of the gloveboxes are open for solution transfers. Dividing the number of hours for the open doors by the total operating hours of 1760 for 10 months (expected operating time per year) yields 1.4E-3, for a probability that the doors would be open during any 1 hour of operation. Assuming a failure rate of 1E-2 for mechanical equipment and multiplying the probability the doors would be open during any 1 hour yields approximately a probability of 1E-5 that the doors would be open during an equipment failure to initiate an accident. This probability is in the extremely unlikely bin for hazards analysis.

Conclusion: The DNFSB letter of April 23, 2002,[1] and the associated Staff Issue Report[2] regarding the new Aqueous Recovery Line for Plutonium-238 (Pu-238) Scrap Recovery at Los Alamos National Laboratory raises some questions regarding hazard analysis, identification of controls, and specification of Technical Safety Requirements. It is the Laboratory position that the Unreviewed Safety Question, the Process Hazard Analysis and supporting documentation, coupled with the Safety Basis Amendment Request approved by the DOE[5] provide a thorough identification and analysis of hazards, comprehensive set of controls, and clear specification of Technical Safety Requirements.

The questions outlined in the Staff Issue Report,[2] while adequately covered by the TA-55 authorization basis, fall directly in line with TA-55 and Laboratory initiatives for continuous quality improvement. Several TA-55 PrHAs prepared over the last five years have been extensively reviewed by both DOE/NNSA and DNFSB staff. While NMT Division believes the PrHAs set the standard for thoroughness in hazard analysis in the NNSA complex today, preparation of the TA-55 Safety Analysis Report (SAR) Upgrade has identified areas of potential improvement that will be used in development of the next generation of PrHAs.

The Laboratory is also committed to continuous quality improvement in the authorization basis arena as evidenced by the formation of the Office of Authorization Basis, formalization of the Unreviewed Safety Question process, development of the Laboratory Implementation Requirements and Guidelines, and placement of authorization basis deliverables in the UC Contract. By formalizing the authorization basis process and teaming with the NNSA, authorization basis coverage of Los Alamos National Laboratory nuclear facilities continues to improve.

References:

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18. ***Technical Report on Hydroxylamine Nitrate,*** DOE-EH-0555, February 1998
19. TA-55 Authorization Agreement, documented in Los Alamos National Laboratory memorandum dated December 22, 1999 from LANL Director John Browne to DOE Area Manager David Gurule'
20. DOE Order 5480.21, *Unreviewed Safety Question*

Enclosure 2

02 - 1368

Identification of Implicit Issues and OLASO Responses

ISSUE 1: DNFSB Staff inquired relative to why NNSA pursued conditions-of-approval on USQ approvals. Further discussion on this occurred in the context of how exactly conditions-of-approval fit into the regulatory framework of 10CFR830 Subpart B, and why they were required and/or allowed from a regulatory basis perspective. Conditions-of-approval for the USQ were specifically alluded to in the letter in the context that the deficiencies in the safety basis proposed by LANL might have been resolved more effectively had DOE held LANL responsible for reanalyzing the potential accident scenarios and developing a complete set of functionally classified safety controls.

NNSA/OLASO RESPONSE:

APPLICABLE REGULATORY GUIDANCE:

DOE G 424.1-1 1 (USQ), Section 1, introduction, pg. 1:
"...The unreviewed safety question (USQ) process is primarily applicable to the documented safety analysis (DSA). Although the rule references only the DSA, the DSA must include **conditions of approval in safety evaluation reports** and facility-specific commitments made in compliance with DOE rules, Orders, or Policies."

DOE G 424.1-1 1 (USQ), Section 2.4, pg. 2:
"If a change is proposed or a condition is discovered that could increase the risk of operating a facility beyond that established in the current safety basis, DOE line management, including, where applicable, the NNSA, **must review and determine the acceptability of that risk through the process of approving a revised safety basis that would be developed and submitted by the contractor**."

DOE G 424.1-1 1 (USQ), Section 2.4, pg. 5:
"If a USQ is determined to be present, the evaluation of the safety of the situation will require **not only DOE's review but also its approval of resulting changes before any operational restrictions are removed.**"

10CFR830.3(a) DEFINITIONS:
"Safety evaluation report means the report prepared by DOE to document.

(1) The sufficiency of the documented safety analysis for a hazard category 1, 2, or 3 DOE nuclear facility;
(2) The extent to which a contractor has satisfied the requirements of Subpart B of this part; and
(3) The basis for approval by DOE of the safety basis for the facility, **including any conditions for approval.**"

10CFR830.202(c)(3):
"In maintaining the safety basis for a hazard category 1, 2, or 3 DOE nuclear facility, the contractor responsible for the facility must:...**Incorporate in the safety basis any changes, conditions, or hazard controls directed by DOE.**"

The above quotes help to clarify and establish the regulatory basis for conditions-of-approval in the context of the USQ process. In particular, a USQ approval (and associated Hazard and Accident Analysis where applicable) is a modification to the facility safety basis and must be formally authorized by DOE. Since DOE approval is required for a USQ, the USQ and the approval (essentially a SER), as well as any conditions-of-approval contained therein become a modification of the safety and authorization bases basis for the facility. As identified in the LANL memorandum, when the Hazard/Accident analysis submitted to DOE is superposed with the NNSA approval memo (and conditions of approval), all hazards were appropriately addressed, a complete set of controls was specified, and the residual risk was determined to be acceptable to the Administration in accordance with its regulatory authority.

It should be noted that conditions-of-approval are never needed in a perfectly completed USQ (and associated Hazard Analysis/Accident Analysis, and derivation of controls where applicable). Rejection of all imperfect USQs/HAs/AAs or authorization bases would result in unacceptable cessation of the program. Therefore, since almost all USQs/HAs/AAs are imperfect, it follows that a judgement determination must, in almost all cases, be made in the context of a balance between program need and consideration of the attendant risks (or difference in acceptability of risk between the contractor side and federal side) as to whether an imperfect USQ/HA/AA submittal should be rejected to correct problems or approved with changes and conditions-of-approval.

ISSUE 2: THE DNFSB Staff Report stated that "LANL plans to start up the scrap recovery process before incorporating many of the safety-related administrative controls—including controls that implement functions designated as safety-class by DOE—into Technical Safety Requirements (TSR). LANL intends to make such changes to the plutonium facility TSR in its 2003 annual update. Many of the DOE-mandated controls warrant the more stringent TSR-level implementation and surveillance requirements before startup of the process." The staff verbally indicated a further concern that the TSRs need to generally be modified to support all USQ approvals.

NNSA/OLASO RESPONSE:

The portion of the concern within the quotes above was fully addressed in the LANL memorandum response and will not be reproduced here. The concern about needing to modify TSRs each time a USQ is approved will be addressed here.

APPLICABLE REGULATORY GUIDANCE:

DOE G 424.1-1 1 (USQ), Appendix B page B-6:
"When the USQ determination is positive, indicating the need for DOE review and approval of the change, the safety analyses and controls associated with the approved action become part of the safety basis for the facility. **Any changes necessary to the DSA and TSR documents as a result of the change should be incorporated at the next annual update.** The results of the USQ determination define the need for DOE approvals of the supporting criticality safety evaluations **and explicit updates of the DSA and TSRs.**"

A USQ may occur when a new proposed process or experiment outside of the approved safety basis is proposed (such as occurred with the proposed Scrap Recovery Process). Once a USQ determines that DOE approval is required to implement the change, a safety basis amendment is required prior to implementation of the change. The safety basis amendment includes supporting hazard and accident analyses, derivation of controls and the Safety Evaluation Report (SER) approval, which may be stand-alone authorization basis documentation from the SAR and TSRs. These controls may be such that no immediate modification to the TSRs is required (e.g. no new LCO, no change to existing LCOs, no change to TSR required surveillances, etc.). Indeed, per DOE G424.1, Section 3.2 (Screening):

"Candidate items **for screening** include situations wherein the USQ process **may not be applicable** (sic such as):

- changes to a requirement in the TSRs, or the addition of a new TSR requirement;
- changes that management has already decided will be submitted to DOE for safety review and approval (including TSR changes, above)..."

Thus, if there is a required change to the TSRs then this is generally handled outside of the USQ process (though some, but not all, TSR changes may be driven by a USQ/HA/AA).

Therefore, as stated previously, DOE conditions of approval are allowed for USQ approvals as modifications to the Safety Basis, USQs can occur that require no TSR changes, and the DOE approval of USQs is itself part of the Safety Basis. Once a USQ change is approved (and associated HA/AA) that approval document, the USQ, the HA/AA are an addendum to the facility Safety Basis and are required to be in effect (verified) prior to operations. Future updates to the Safety Basis then capture these addenda in the DSA/TSRs. In point of fact, documented analyses of proposed new processes are often submitted with the USQ. If these documented analyses involve passive safety controls then no change to TSRs (e.g. LCOs, surveillances, or basis statement changes, etc.) may be required. Sub-tier documents like procedures that are called into existence by the HA/AA or DOE approval may provide adequate Administrative Controls until the Safety Basis is updated on an annual frequency as applicable.

To further strengthen this logic, one may consider the opposite case and hypothesize that every USQ approval needs to result in a TSR amendment and see where this assumption logically leads. In a moderately complex facility, engaged in cutting edge research and experimentation, there may be 10 to 20 or so USQ approvals in a year. Under the initial assumption of a TSR modification for each USQ approved, this would translate into a TSR modification about every two to five weeks for the facility. If there are 20 nuclear facilities on the site (as there is approximately at LANL), then this could translate into a TSR modification and re-approval on an approximate daily or weekly basis. This is clearly untenable, and one may infer that the CFR and Guides are prudent in their approach to handling USQs in not necessitating a TSR mod for every USQ approval. With specific regard to the Scrap Recovery USQ, the safety systems named were either passive controls, already existing controls, defense-in-depth controls, or safety controls controllable through sub-tier procedures and so fit the paradigm of not requiring an immediate TSR amendment. Therefore the Scrap Recovery USQ could (and was) approved without modification to the TSRs.

ISSUE 3: The DNFSB Staff Report discusses issues relating to *level of controls* (Safety Class (SC), Safety Significant (SS), Defense-in-Depth (DinD), or Administrative Controls (ACs)). In the context of discussions with the staff relating to the Scrap Recovery Process, (specifically the wire mesh around the ion exchange column mentioned in the Staff Report) interactions occurred where it was stated that Safety Significant Structures, Systems, and Components (SS-SSCs) were support for Safety Class. NNSA/OLASO took this to be one of the implicit issues in the DNFSB letter to be addressed per Staff guidance on the response to the letter.

NNSA/OLASO RESPONSE:

LANL addressed the explicit issue above in Attachment 1 but it was not clear that LANL knew of the implicit issues and guidance supplied by DNFSB Staff relative to the specific issue of the wire mesh being a support system for safety class, or the generic issue of SS-SSCs being support systems for SC-SSCs. Therefore this specific issue will be addressed by NNSA.

APPLICABLE REGULATORY GUIDANCE:

DOE G 421.1-2 (SAR), pg. 31, 5.3.2 SELECTION PROCESS:
"DOE-STD-3009-94, Change Notice No. 1, **invokes safety significant SSCs for either defense in depth or for worker safety.** No dose criteria are assigned as EGs in these cases. The classification of SSCs as safety **significant should be based on qualitative assessments. In the case of defense in depth, SSCs designated as safety significant are selected to prevent or mitigate accidents of lesser consequence or to provide extra layers of protection beyond that provided by safety class SSCs.** The specifics of safety significant designation intent are addressed in the definitions section of DOE-STD-3009-94, Change Notice No. 1 or successor document, in the Introduction, and in Sections 3.3.2.3.2, Defense in Depth, and 3.3.2.3.3, Worker Safety."

DOE G 423.1-1 (TSR), pg. 15, Section 4.15 Safety Structures, Systems, and Components:
"...**Support systems for Safety Class SSCs would normally be considered to be Safety Class if they are relied upon to support a safety class function...Support systems for Safety-Significant SSCs should be considered safety significant.**"

This and many other regulatory references unequivocally establish that safety significant SSCs are qualitatively named based upon engineering judgement for:

1. Major contributors to defense-in-depth (prevent or mitigate accidents of lesser consequence than those that challenge the EG),
2. Barriers that prevent acute or serious injury to workers,
3. By the quotes above and the TSR General Operability Requirements, support SSCs *required* to ensure the operability of SC-SSCs are themselves SC-SSCs and support SSCs *required* to ensure the operability of SS-SSCs are themselves SS-SSCs.