CNWRA 91-015

# SAFETY EVALUATION REPORT ON THE WEST VALLEY DEMONSTRATION PROJECT SUPERNATANT TREATMENT SYSTEM

**A Review of**
**Safety Analysis Report, Volume III, Part D**
**(WVNS SAR-004 Rev. 6)**

*Prepared for*

**Nuclear Regulatory Commission**
**Contract NRC-02-88-005**

*Prepared by*

**Center for Nuclear Waste Regulatory Analyses**
**San Antonio, Texas**

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**November 1991**

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# 1 INTRODUCTION

## 1.1 DESCRIPTION OF THE WEST VALLEY DEMONSTRATION PROJECT

The West Valley Demonstration Project (WVDP) is being undertaken by the Department of Energy (DOE) pursuant to the West Valley Demonstration Project Act (P.L. 96-368) (Ref. 1). This Act directs the WVDP to five major activities, as follows:

- Solidify the liquid high-level radioactive waste (HLW) stored at the site;
- Develop containers for the solidified HLW;
- Transport the waste to a federal repository for disposal;
- Dispose of low-level radioactive waste (LLW) and transuranic (TRU) waste produced during the project; and
- Decontaminate and decommission facilities.

It is the task of WVDP to convert the high-level radioactive waste (HLW) into borosilicate glass for ultimate disposal in a federal repository. The WVDP operations are controlled by DOE directives, with oversight provided by NRC to ensure no significant risk to public radiological health and safety. The WVDP site is located on New York state's western plateau, near the community of West Valley, about 55 km south of Buffalo in the town of Ashford, Cattaraugus County.

The West Valley site was a commercial nuclear fuel processing center that operated from 1966 to 1972. The DOE, together with its prime contractor, Westinghouse Electric Corporation, officially took over the site operation in February of 1982. The DOE was mandated by Congress to carry out a HLW management demonstration project at this site. This project involves the cleanup of over 2.1 million liters of nuclear spent fuel reprocessing wastes. The terminal form of the HLW will be borosilicate glass. A waste characterization program was immediately set in place to provide necessary input into several process design activities. As a result, it was determined that the HLW is stored in two tanks containing two different waste forms. Alkaline Plutonium Resin Extraction (PUREX) wastes are stored in Tank 8D-2 (approximately 2.1 million liters), and acidic Thorium Resin Extraction (THOREX) wastes are stored in Tank 8D-4 (approximately 31,000 liters). The THOREX wastes exist in essentially a single, liquid phase, but the PUREX wastes exist in two distinct phases, referred to as the supernatant (the liquid phase) and the sludge (the solid phase).

To perform the activities mandated by the Act, several supporting systems will be needed. Two of the most important of these systems are the Supernatant Treatment System (STS) and the Sludge Mobilization and Wash System (SMWS). The primary objective of the STS is to reduce the volume of liquid HLW, while the primary purpose of the SMWS is to remove sodium sulfate salts which would be detrimental to the glass waste form.

The pretreatment of HLW before solidification is performed by the SMWS and the Integrated Radwaste Treatment System (IRTS) that includes: STS as well as the Liquid Waste Treatment System

(LWTS), the Cement Solidification System (CSS), and the Drum Cell (DC). The DOE has proposed different Safety Analysis Reports (SARs) that describe and analyze each system, shown in Figure 1. Figure 2 is a schematic of the process that will be used to transform the HLW to the borosilicate glass form. This Safety Evaluation Report (SER) focuses only on the STS system described in Revision 6 of WVDP SAR-004 (Ref. 2). Since the SMWS has been incorporated to the scope of the SAR, the scope of this SER includes the SMWS. As used in this SER, "SAR" refers to Revision 6 of the WVDP SAR-004 (Ref. 2), unless otherwise specifically stated.

## 1.2 PURPOSE AND SCOPE OF THE SAFETY EVALUATION REPORT

This SER addresses the public radiological health and safety impacts of operating the STS. The SMWS is also incorporated in the scope of this SER to the extent that the SMWS relates to the STS as described in the SAR on the STS. The primary source of information for this evaluation and review was the DOE SAR entitled "West Valley Demonstration Project Safety Analysis Report, Volume III, Part D, Supernatant Treatment System, Rev. 6" (Ref. 2). This SAR describes STS and SMWS facilities design, focusing on the structures and equipment that are used and their layout. The functions of the two systems are discussed in the SAR for normal, abnormal, and accident conditions. Also, waste confinement and associated management criteria are discussed, including the process itself. Additional information was obtained during on-site visits to WVDP, and from other documents as referenced in this SER.

In the following sections, an overview of SMWS and STS equipment, configurations, and operating procedures is presented. Also, an independent evaluation of the off-site dose releases to the public was conducted, by comparison of the estimated release levels to applicable standards, including the Code of Federal Regulations (CFR), Title 10 (Energy), Chapter I (NRC), Part 20, dated May 1991 (Ref. 3). The SAR (Ref. 2) and its reference documents were thoroughly reviewed for completeness and technical adequacy.

## 1.3 PROCEDURE FOR DEVELOPING THE SER FOR REVISION 6

Safety Evaluation Reports (SERs) have previously been written with respect to the prior revisions of the SAR on the Supernatant Treatment System (STS) (Ref. 4) and the Liquid Waste Treatment System (LWTS) (Ref. 5). These SARs are presented in Volumes III and IV (Refs. 2, 6), respectively, of the DOE's SAR for the entire project. The SAR Volume III (concerning the Supernatant Treatment System) has again been significantly revised since the issuance of an SER on the STS in September, 1987. The revision was necessary to expand the scope of the analysis to include the Sludge Mobilization Wash System (SMWS). Also, the modification incorporated comments submitted by DOE's Technical Review Group for Safety Analysis Reports and reflected a change in operational strategy after the discovery of excess plutonium passing through the STS in late 1990. This SER is intended to evaluate concerns raised as a result of these changes and also unresolved concerns expressed in the previous STS SER.

In developing the SER, a detailed examination was conducted of various technical aspects presented in the SAR and associated comments presented by the DOE's Technical Review Board during the development of the final SAR. During the examination, opportunities were provided to attend DOE presentations and seek clarification of several safety issues. DOE has responded to the several clarifications and other comments and incorporated most of them into the final SAR.

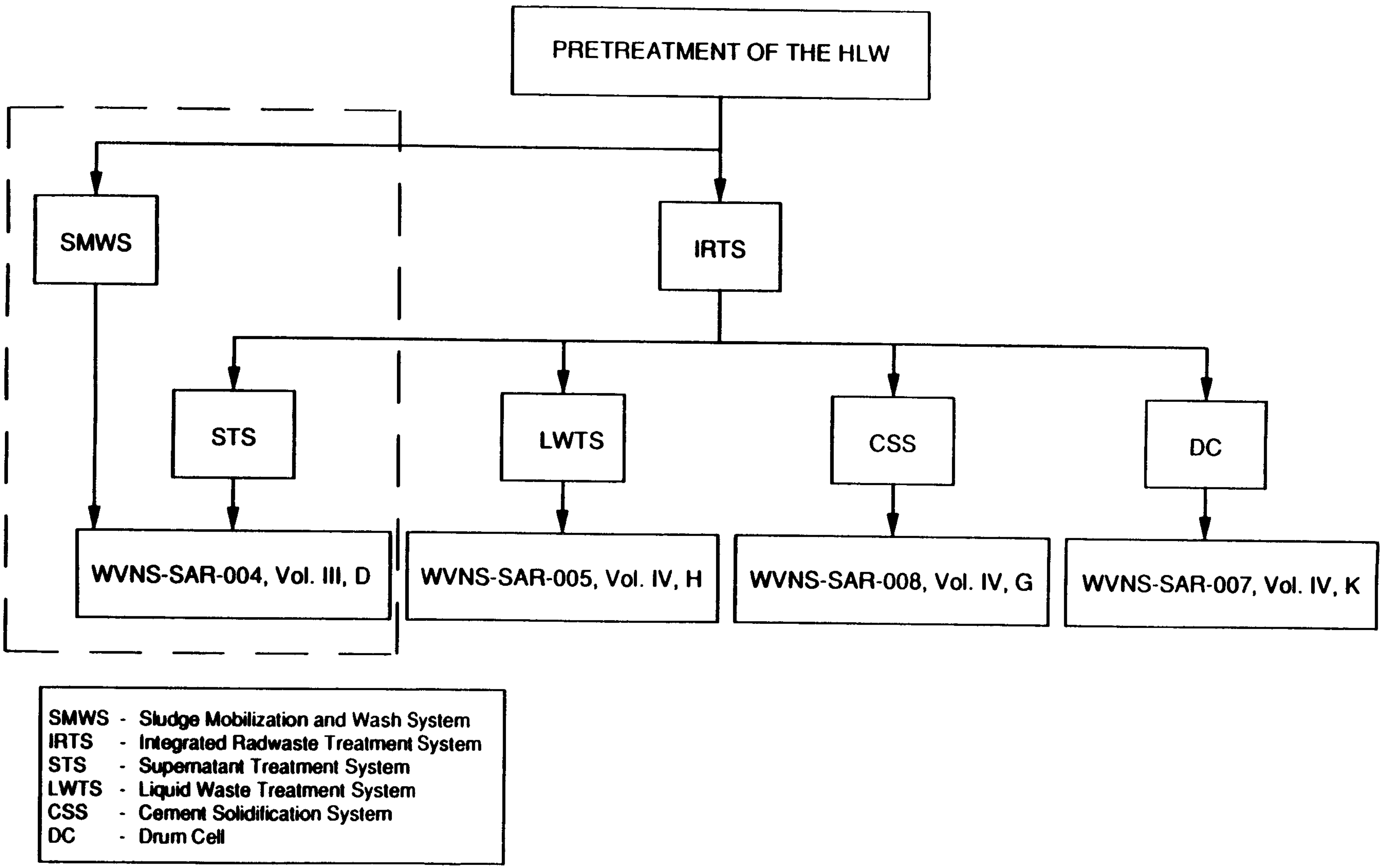



Figure 1. Pretreatment Systems at WVNS and Their Related Safety Analysis Reports (SARs)

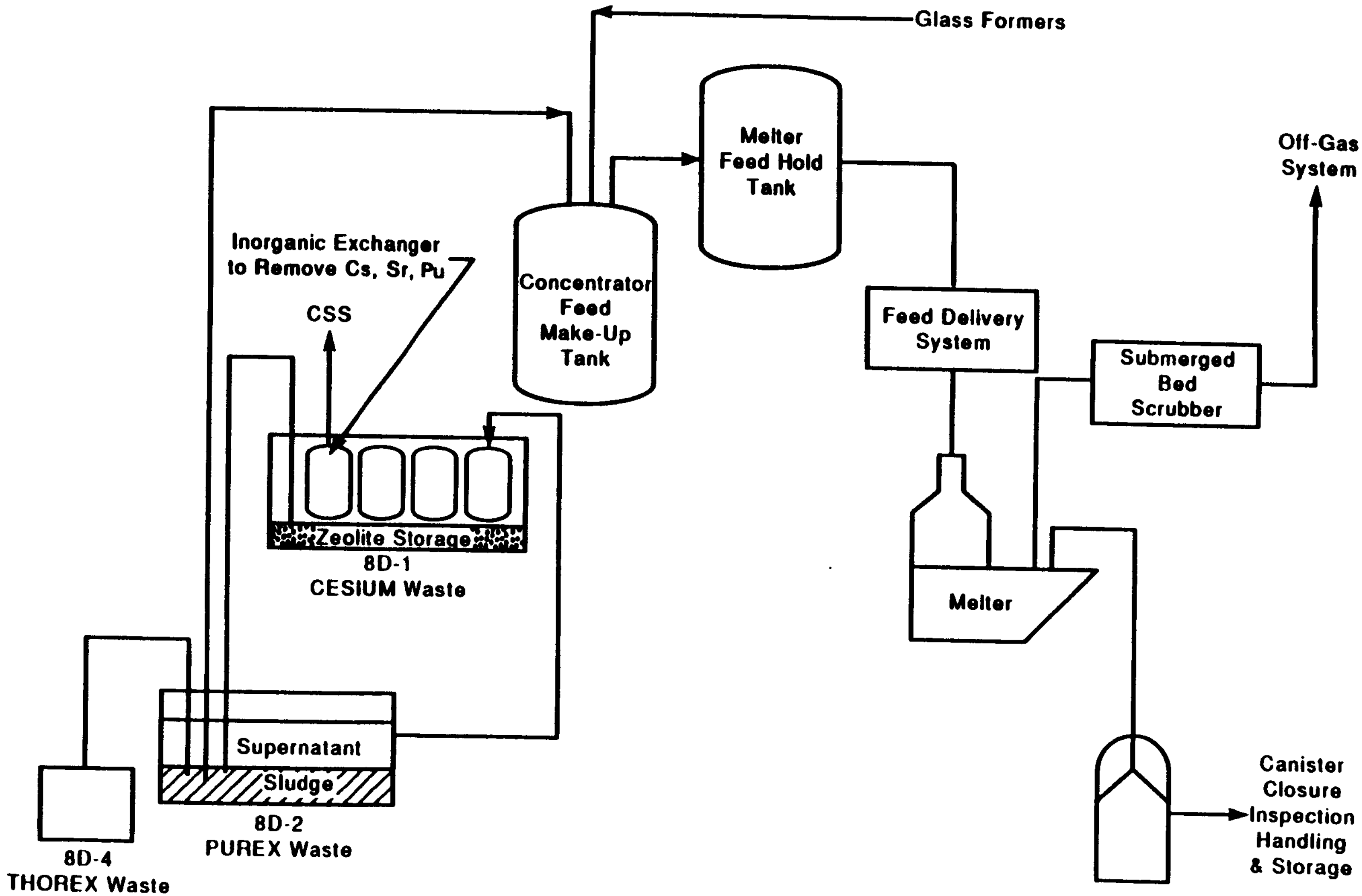



Figure 2. High-Level Waste Processing Flow Diagram

## 1.4 EXECUTIVE SUMMARY OF CONCLUSIONS

The safety evaluation of changes which have been incorporated into the STS concludes that the Sludge Mobilization and Wash System will be operated within controls established to protect public radiological health and safety. This conclusion is based on the following expectations:

(1) Examination of the gaseous effluents from the sludge washing and wash treatment normal operations reveals that the site boundary dose is far below allowable limits. The calculated maximum individual off-site effective dose equivalent for all radionuclides is less than 70 $\mu$rem (0.7 $\mu$Sv) annually.

(2) Examination of the potential for and effects of abnormal operations and accidents from both externally and internally caused events reveals that site boundary dose is small. The most severe credible accident of a fire in a Waste Tank Farm Ventilation System (WTFVS) High Efficiency Particulate Air (HEPA) filter would result in a 0.5 mrem (5 $\mu$Sv) committed effective dose equivalent (CEDE) at the nearest residence.

(3) The experience gained through operation of the STS since 1988 for the removal of cesium from the supernatant can be applied to the continued operation of the system for removal of the sodium sulfate salts. The solubility of the plutonium (Pu) and strontium (Sr) in the wash solution is an important parameter for control. The use of titanium-coated zeolite to remove the soluble transuranics in order to meet NRC's requirements for stable low-level radioactive waste form disposal is contingent on the levels of soluble Pu measured in the wash solution. The NRC expects DOE to accurately measure, record, and report the Pu levels in the wash solution and STS treated water, and to ascertain when its removal becomes necessary to prevent transport to the LWTS.

(4) Additional efforts are justified to monitor the potential for the three major types of corrosion (wall thinning, concentrated pitting, and stress corrosion cracking) on the carbon steel tank material.

# 2 THE SUPERNATANT TREATMENT SYSTEM

## 2.1 DESCRIPTION OF THE PHYSICAL SYSTEM

The principal components of the STS are associated with the Tank 8D-1. In addition to support hardware and instrumentation, they consist of the zeolite (ion exchange) columns, supernatant feed tank, sluice feed tank, supernatant cooler, prefilter, postfilter, and sluice water feed pump. Figure 3 shows some of these components and their locations within the STS. The SMWS components include the tank 8D-2 and the five (5) sludge mobilization pumps. The arrangement of the STS components is shown in Figure 4.

In support of both the STS and SMWS are the WTFVS and its associated HEPA filters, the STS support building which contains the control room, the valve aisle which contains instrumentation and valves, and piping runs. The Permanent Ventilation System (PVS) is routinely used for ventilation of components other than Tanks 8D-1 and 8D-2, and it can be used in the event of failure of the WTFVS. By use of either the WTFVS or the PVS, the HLW tanks are maintained under negative air pressure (0.5 inches Waste Column [WC]), so that airborne contaminants are drawn by ventilation through the HEPA filters before the filtered air is exhausted through the stack. Also, there is a positive hydraulic head maintained in the gravel backfill surrounding the foundation of the tanks, so that, if one of the tanks ruptured, and the concrete vault surrounding the tank cracked or failed, the result would be flow of the surrounding water into the tank rather than flow of sludge or untreated water into the surrounding gravel backfill. Figure 5 shows the 8D-2 Tank with the sludge mobilization pumps in place and supported by the truss structure.

The following safety protection systems/features are identified in the SAR: 1) multiple confinement barriers and systems for STS and SMWS (primary and support barriers to releases are listed in Table D4.3-1A of the SAR); 2) protection by equipment and instrument design and selection; 3) criticality safety; 4) radiological protection; 5) fire and explosion protection; and 6) radioactive waste handling and storage.

## 2.2 THE STS PROCESS

The HLW which is stored in Tank 8D-2 is from former PUREX reprocessing and has separated into an alkaline supernatant and sludge. The STS process is described in conjunction with the SMWS, since the scope of the SAR on the STS has been expanded to include the SMWS. The supernatant is supplied to the STS after washing to resolubilize salts through agitation by the sludge mobilization pumps. The operation of the STS is contingent on effective mobilization and mixing of the sludge by the pumps.

The purpose of the SMWS is to remove unwanted sulfate salts from the sludge, as these would be detrimental to the final glass waste form. The SMWS will suppress actinide solubility while dissolving the salt and salt crystals for final removal through the LWTS. This will be done in a batch process, which is expected to be repeated for a total of four cycles. A dilute caustic solution is to be introduced to the 8D-2 tank containing the sludge. A high pH level is needed to prevent transportation of plutonium after sludge mobilization and during wash solution treatment. Additional water will be added to Tank 8D-2 to mix with the sludge so that interstitial salts will be dissolved and transported with the wash solution for subsequent solidification to produce a solid LLW in the form of cement within square,

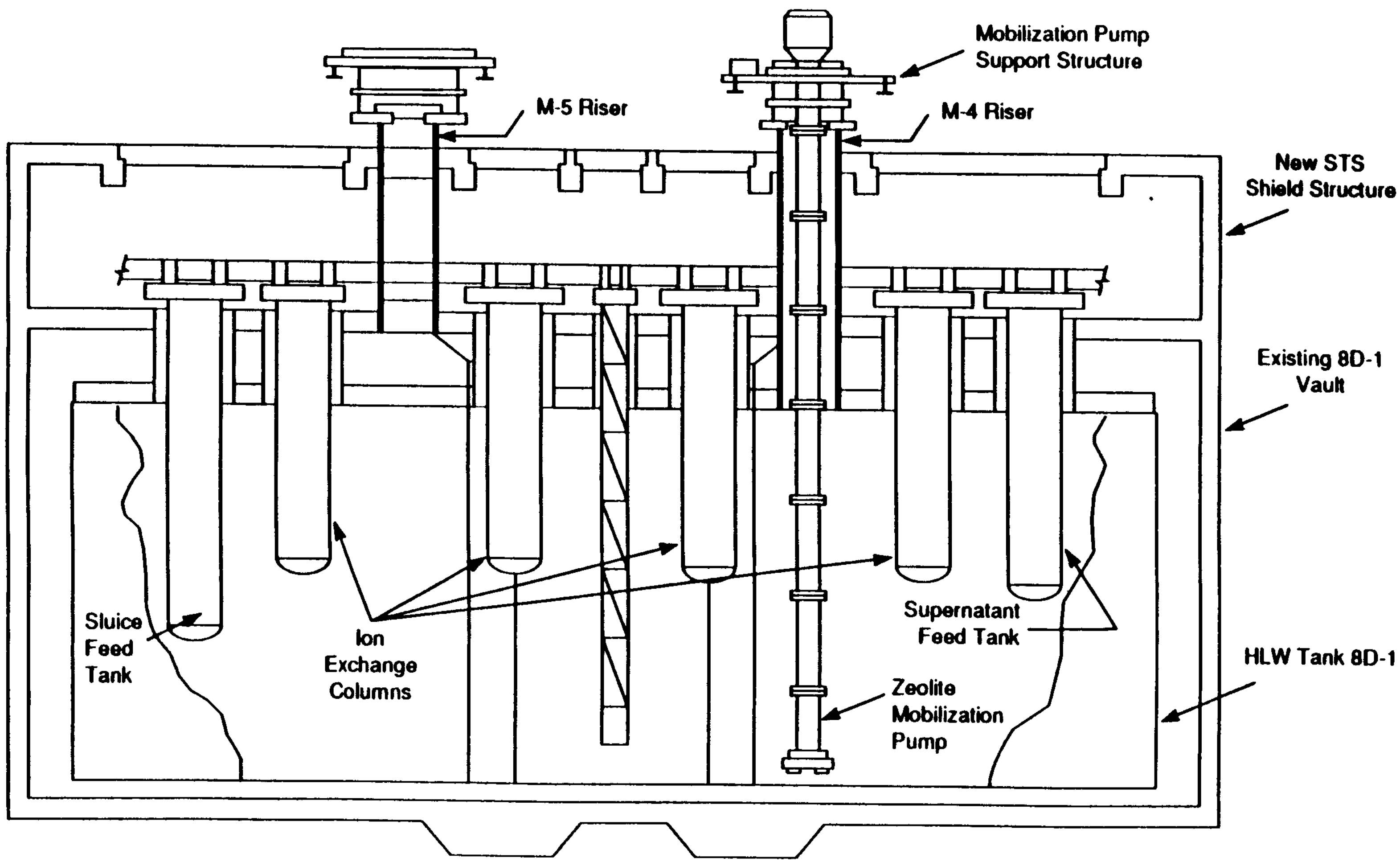
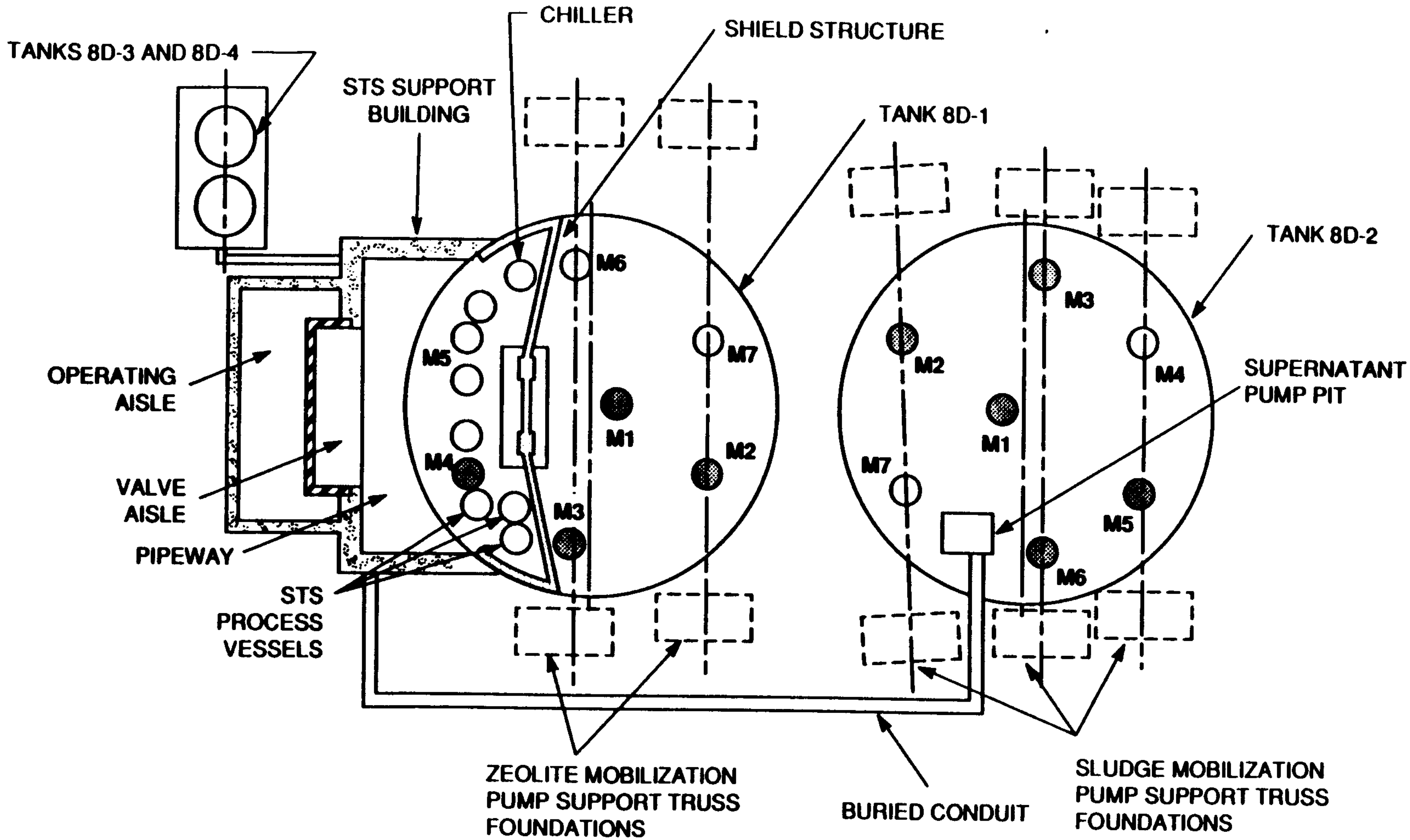



Figure 3. Tank 8D-1 and Internal Supernatant Treatment System Components




| TANK | RISER NUMBER | PUMP FUNCTION |
|---|---|---|
| 8D-1 | M1 TO M7 | SPENT ZEOLITE MOBILIZATION |
| 8D-2 | M1 TO M7 | SLUDGE MOBILIZATION AND WASH |

**Figure 4. Arrangement of Supernatant Treatment System and Sludge Mobilization and Wash System Components**

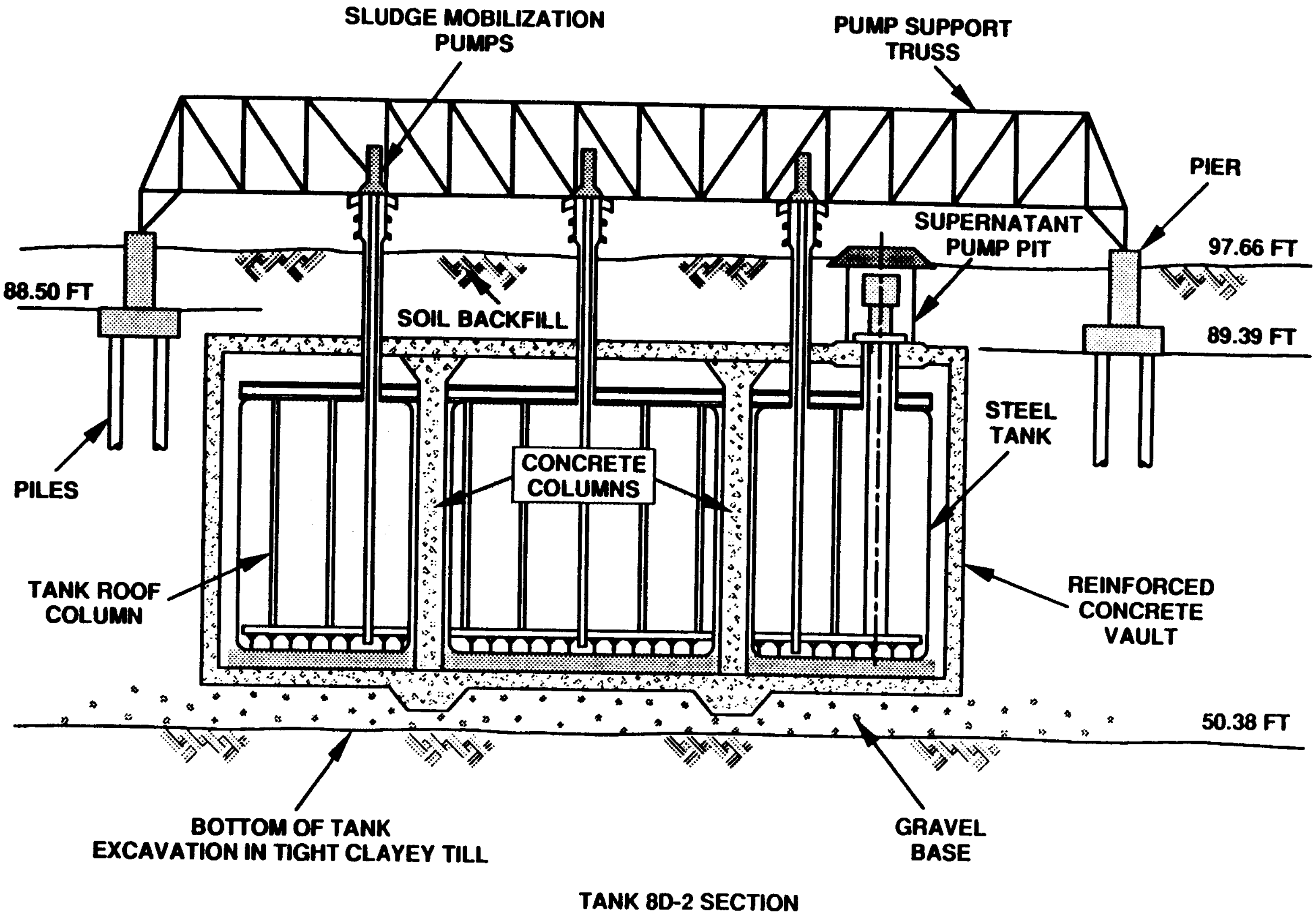



Figure 5. Elements of the Sludge Mobilization Wash System

71-gallon drums. The sludge will be mobilized into the wash solution by special sludge mobilization pumps which circulate and agitate the mixture within Tank 8D-2. Figure 6 shows the locations where the pumps will be inserted to agitate the sludge. During this process, the unwanted salts will be dissolved and remain in the supernatant, below which the sludge again settles, allowing supernatant to be filtered and drawn off and processed in the STS. After completion of an anticipated four cycles of the sludge mobilization and wash process, it is expected that the concentration of sulfate salt in the remaining sludge will be acceptable for producing a glass waste form. If not, additional cycles of the wash process will be performed until the salt concentration is acceptable. The washed sludge will remain in Tank 8D-2 until it is used as part of the feed to the Vitrification Process (VP).

The STS has decontaminated up to 99% of the treated alkaline supernatant, specifically by removal of 80% of cesium-137 ($^{137}Cs$) from the original concentration of untreated supernatant. This was achieved through a floating suction pump, a prefilter, a chiller/cooler, a series of four zeolite (ion exchange) columns, and a post filter, all located in Tank 8D-1. The ion exchange medium, contained within the zeolite columns, is zeolite-based. Three of the four columns will be active at a given time, so that the fourth can be flushed and recharged with zeolite. A titanium-coated zeolite has been developed for use in one or more of the zeolite columns for removal of soluble plutonium from the supernatant, if the measured levels of plutonium in wash solution samples are above specified levels. The chiller, a brine shell-and-tube heat exchanger, cools the supernatant before delivery to the zeolite columns, because the ion exchange process is more efficient at lower temperatures. The STS can be operated in either batch mode or continuously.

The STS process creates stable LLW as a saline supernatant. There are three kinds of liquid that are processed in the STS. One is liquid from Tank 8D-2 that has passed through the STS ion exchange system and is subsequently a decontaminated supernatant. Another is the decontaminated supernatant liquid plus flush water that is used as a final step for STS operation. In this flushing operation, residual supernatant is pumped through the zeolite columns using either water or liquid from Tank 8D-1 before processing by the STS. Typically 10,000 gallons of Tank 8D-1 liquid or 6,000 gallons of water would be used to flush the columns. The third kind of liquid processed in the STS is the liquid from Tank 8D-1 that consists of (1) water condensed from vapor generated in Tank 8D-2, and (2) water generated from various STS operations. This third kind of liquid has a relatively low salt concentration but a constantly increasing cesium concentration, since Cs-loaded zeolite has been and is continuously stored in Tank 8D-1.

After removal of cesium (decontaminated by a factor of 1000) and possibly plutonium in the STS, the salt-laden wash solution is passed on to the LWTS, which is used to produce a solid LLW in the form of cement within square, 71-gallon drums. These drums are stored on-site, awaiting ultimate LLW disposal.

## 2.3 ADMINISTRATIVE CONTROLS

The protection of the radiological health and safety of the public is provided through the administrative controls that have been established and practiced at WVDP. The WVDP administrative controls that are important to safety issues are the West Valley Nuclear Services Co., Inc. (WVNS) procedures that are prepared, implemented, reviewed, maintained, and approved as directed by the WVNS Policy and Procedures Manual (Ref. 7) in accordance with DOE Orders, directives, regulations, and guidance. The WVNS procedures control all project activities from the design through the operational phases.

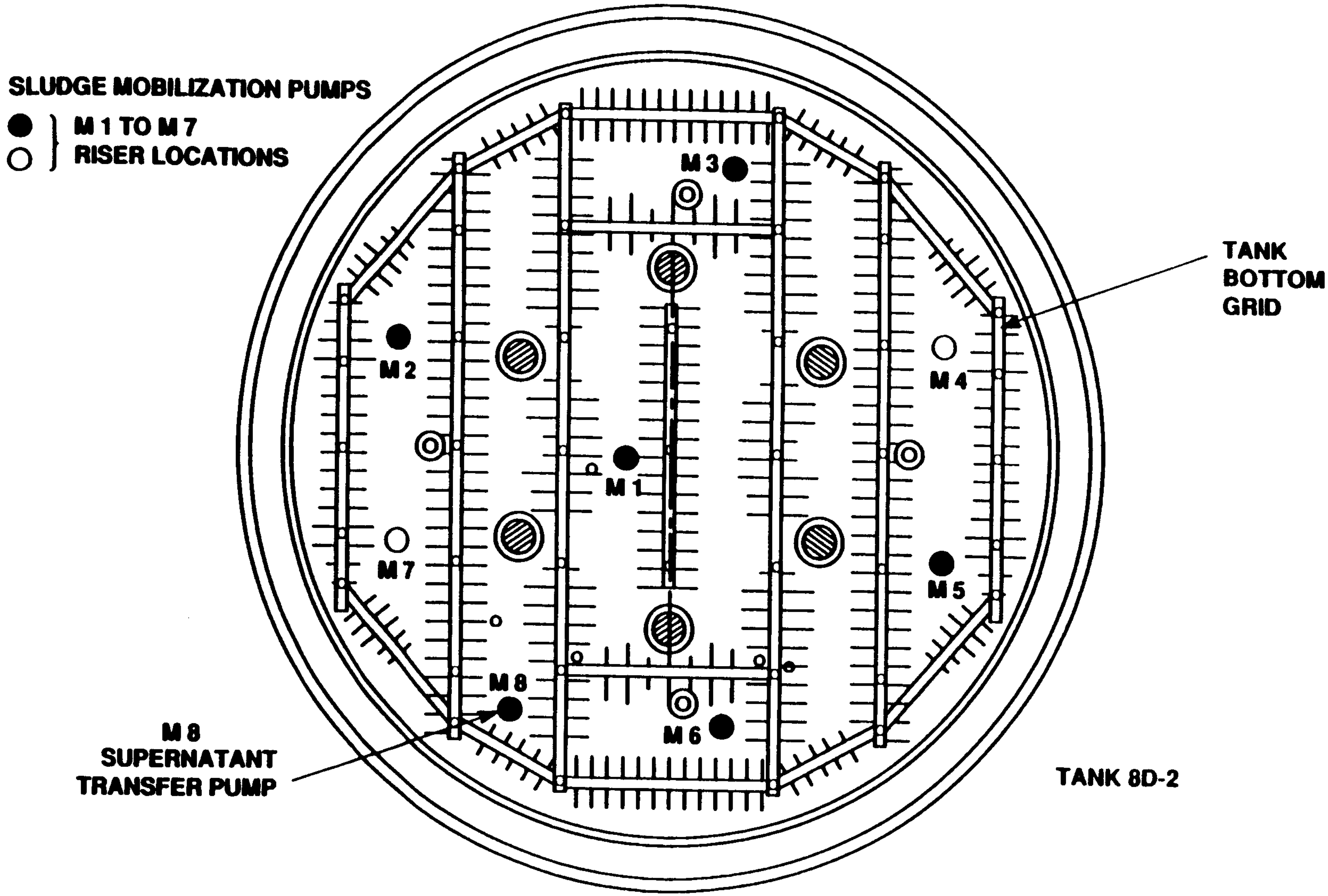



Figure 6. Tank 8D-2 Sludge Mobilization Pumps and Supernatant Transfer Pump Locations

The specific administrative procedures for operation of the STS and SMWS are described in Chapter 10 of Volume IV of the SAR (Ref. 6). This chapter includes requirements for pre-operational testing and operating procedures, for operator training and qualification programs, for normal operating procedures, and for emergency response planning.

Quality assurance (QA) for the WVDP is discussed in Volume I of the SAR (Ref. 8, Section A.12.0). With respect to review of structures, systems, and components important to safety, a discussion is included in Section A.4.4 of Reference 8. NRC periodically performs independent QA audits for the WVDP processes and activities to verify that appropriate QA controls are in place.

# 3 PERFORMANCE EVALUATION

This section provides results of the evaluation conducted on the SAR. The NRC staff with the assistance of the Center for Nuclear Waste Regulatory Analyses (CNWRA) has reviewed the revised SAR on the STS, as well as other volumes of the SAR (Refs. 6, 8) which have been previously issued. NRC's previously prepared SER (Ref. 4) has also been reviewed to identify possible points of concern which remain unresolved in the SAR. In addition, new items identified in this SER include those arising as a result of (1) new information in the most recent STS SAR; (2) expansion of the scope of the most recent STS SAR to include the SMWS; and (3) new information in the latest revision of 10 CFR Part 20 (Ref. 3).

The staff has reviewed the STS and SMWS for the potential of the release of radioactive substances to the environment. This section contains an evaluation of radionuclide release rates and impacts on public health and safety for normal operations, abnormal operations, and accidents. The evaluation of impacts for normal operations considers both the STS operations and the SMWS operations since the performance of each is intimately connected with that of the other, although the two operations will be performed separately. The evaluation of abnormal operations and accidents considers both internally-initiated and natural events including lightning, earthquakes, tornados, criticality, fires, and explosions.

## 3.1 NORMAL OPERATIONS

The SMWS, under normal operations, involves the mobilization or agitation of the sludge inside Tank 8D-2. A sectional view is given in Figure 5. The sludge wash/supernatant solution is simply drawn into the SMWS pumps and discharged through two pump casing nozzles back into the Tank 8D-2 volume. The agitation period is expected to be about five days per pump operation. After each of the expected four SMWS pump operations, the sludge is allowed to resettle before the supernatant is processed through the STS equipment which is located in Tank 8D-1. There is no liquid discharge to the environment directly from the SMWS or the STS. The liquid effluents from STS/SMWS operation are treated in the LWTS which is discussed in WVNS-SAR-005, Volume IV, Part H (Ref. 6), and reviewed in its corresponding SER (Ref. 5). Releases expected from the ventilation systems during STS/SMWS operation are in concurrence with the regulations of 10 CFR Part 61 (Ref. 10). Two ventilation systems are involved, the Waste Tank Farm Ventilation System (WTFVS) and the Permanent Ventilation System (PVS). The WTFVS normally processes gases coming from Tanks 8D-1 through 8D-4. (Tanks 8D-3 and 8D-4 are not within the scope of this SER.) Off-gases from the tanks along with the off-gas from the STS processing components suspended in Tank 8D-1 are treated and exhausted at the Main Processing Plant Stack. The maximum anticipated off-gas releases do not exceed the capacity of the WTFVS (150 SCFM). In conjunction with the PVS, which can be used to back up the WTFVS to provide additional safety, this condition is deemed acceptable.

The PVS exhausts air for contamination control from the STS valve aisle and pipeway through the PVS stack which is located near Tank 8D-1. The capacity of the PVS is 4000 SCFM, which by far exceeds the anticipated ventilation requirement during pump installation (550 SCFM), as well as the measured value of 1600 SCFM during pump installation. The capacity is, therefore, sufficient to ensure adequate contamination control.

During sludge mobilization pump installation when the tank risers are unplugged and opened, the PVS will be coupled to the WTFVS to draw more air into the openings so that air flow is into Tank 8D-2 and outward into the off-gas treatment equipment. This ensures a safe operating condition for workers.

The total annual maximum off-site effective dose limit has been set at 10 mrem (0.1 mSv) for normal operations only, by DOE Order No. 5400.5 (Ref. 9), in accordance with 10 CFR Part 61 (Ref. 10). (The set value is more conservative than the values given in the Part 61 Regulation.) However, no limit for the West Valley Project is currently in effect for abnormal operations or accidents. Presently the value for this limit is being considered by DOE based on the probability of the accident taking place and the possible extent of the release. This value is expected to be set in the near future by DOE. The calculated maximum individual off-site effective dose equivalent for all radionuclides for normal operations is less than 70 $\mu$rem (0.7 $\mu$Sv) annually, which is much less than the set dose limits.

All exhaust air is sampled continuously and simultaneously through a particulate alpha monitor and a particulate beta monitor. The continuous air monitors provide alarm indications in the STS building control room. The sample media is screened weekly for gross radioactivity, and composites of the filters are analyzed quarterly for specific isotopes, including gamma-emitting radionuclides, strontium-90, and transuranics. These procedures ensure that radioactive releases through the ventilation system are adequately controlled.

Solid wastes expected from the SMWS operations will be (1) filtration media from roughing filters and HEPA filters and (2) the sludge mobilization pumps. These items are controlled through the WVDP-010 and WVDP-019 procedures for Radiological Controls and Long-term Radioactive Waste Management (Refs. 11, 12), respectively. Disposal of these items will be in accordance with the decontamination and decommissioning plan, which is discussed in Sec. 4.

The SAR states that the mobilization pump seal performance will be routinely monitored for leakage. Since any leakage of the sludge along the shaft will result in contamination of the pump, the manner and frequency of the monitoring process should be clearly described. The mechanical seal design is generally very reliable, but the length of the pump shaft is unique so careful observation of the seal performance is warranted. A mitigating factor pointed out in section 2.0 of the "8D-2 Sludge Mobilization Wash System Integrity Review" (Ref. 13) is that internal fluid pressure in the mobilization pump column serves as a barrier against migration of high-level liquid wastes through the lower mechanical seal and pump column flanges and up the pump drive-shaft to the shaft upper mechanical seal. The pressurization system is described and the range of pressure level is given as 40 to 55 psig in the SOP 55-07, "Sludge Mobilization Pump Operating Procedure" (Ref. 14). No radiological release is expected from pump performance.

In Section D.5.2.1.2, "Modification to Tanks 8D-2 and 8D-3," the SAR briefly refers to installation of sludge mobilization pumps in Tank 8D-2 by remote means. However, the step-wise installation process, whereby the length of spacers beneath the pumps is reduced as sludge is progressively mobilized with pump operation, is not evident from the text. The exposure to workers due to pump installation will, therefore, be greater than what it would be if installation were a one-step process, although exposure during removal of spacers is much less than during initial installation of the pump in the riser. Scale model tests (Ref. 15), along with similar operations at other DOE facilities, demonstrated the feasibility of sludge mobilization in Tank 8D-2 as well as zeolite mobilization for Tank 8D-1. The step-wise installation of SMWS pumps should be evaluated in the ALARA program, which is the subject of Section 3.3 of the SAR.

In SAR Table D.5.4-1, "STS Utilities Requirements," instrument air pressure is given as 105 psig. This is high for instrument air pressure as it is used in typical industrial applications, which is more typically near 15 psig. In Figure D.6.1.1, for example, which shows one of the Process and Instrument Diagrams, three levels of instrument air pressure are given. Instrument air up to 100 psig is used in one application to discharge ion exchange material from the zeolite columns. Compressed air for these applications is called "instrument air" because of its pre-conditioning and not because of its pressure level or usage. Administrative and quality assurance procedures are used to prevent inadvertent use of components in instrument air circuits at pressure levels beyond component ratings. (Ref. 8, pp. 151-154.)

### 3.1.1 Corrosion

Corrosion of the carbon steel Tanks 8D-1 and 8D-2 is a concern and is addressed in the SAR. Corrosion can occur in several ways, such as general wall thinning, concentrated pitting, and stress corrosion cracking. While corrosion does not appear to be have been significant in the past, the new environment in the tank from addition of caustic, the heat input of the pumps, and the re-solution of the salts may result in enhanced corrosion. The SAR discussion is not clear on how the different types of corrosion will be monitored and gauged during the remaining tank service lifetime. For instance, it is reported that stress corrosion cracking has never been observed in corrosion coupons that have been removed from either Tank 8D-1 or 8D-2. However, the specific type of coupons and the degree to which corrosion coupons are stressed are important and should be reported along with the qualitative results. Although corrosion of the tanks does not currently and is not expected to in the future present a problem with respect to radiological release, additional efforts to monitor the potential for corrosion are justified to assure the future safety of the tanks.

A quantitative comparison of measured general corrosion (wall thinning) is given in the SAR to show that this type of corrosion is not a problem. Similar quantitative results should be given to indicate that pitting will not be a problem. The NRC staff feels that additional efforts are justified to monitor the potential for the three major types of corrosion (wall thinning, concentrated pitting, and stress corrosion cracking) on the carbon steel tank material.

The section on Corrosion in the SAR deals with containment barriers, the tanks within which either the waste is stored or components of the STS are located. Of most concern are the carbon steel HLW Tanks 8D-1 and 8D-2. Corrosion of the chiller would not breach a primary containment barrier, since the chiller is located within one of the tanks (Tank 8D-1). Leakage of HLW from the chiller would thus be contained by Tank 8D-1. The pressure in the brine loop of the heat exchanger is maintained at a higher level than that of the supernatant loop, so internal leakage in the chiller would result in brine leaking into supernatant. This would cause a pressure loss in the brine loop, which would initiate an alarm. Also, the local area radiation monitor in the vicinity of the chiller would initiate an alarm. The chiller (and its associated freon refrigeration cooler) could be isolated, repaired, decontaminated, or replaced with minimum radiological effect on operating personnel. Leakage of HLW from the supernatant loop to the brine loop is considered highly unlikely since two independent failures would be required: the structural barrier provided by the heat exchanger between the sludge wash solution and salt solution must be breached, and the pressure gradient maintained between these two fluids, which would normally force salt solution into the sludge wash solution (instead of vice versa), must become reversed. Two other potentially mitigating factors are: 1) the brine loop is a closed loop, which could be repaired if it failed independently of the supernatant (or sludge wash solution) loop, and 2) the chiller is only used during STS operation, not continuously.

### 3.1.2 Pre-Operational Testing

Since the STS equipment has been in operation from May 1988, it is unclear which equipment and physical facilities will require pre-operational testing to ensure proper installation and operation. Credit has been taken for the successful testing and operations already complete, and the pre-operational testing for newly installed equipment has been approved in a formal readiness review conducted by DOE prior to SMWS startup. Some equipment used for radiation control that will require testing, monitoring, and maintenance are the radiation monitoring systems. Although more specific information on the frequency of operations testing and maintenance of these systems could be provided in the STS SAR, Volume I of the SAR, in Section A.8, describes calibration of radiation monitoring systems.

In SAR Section D.10.2.3.10, "Ion Exchange Columns," the pre-operational test acceptance criteria for the ion exchange columns include an extraction efficiency of 99.9% of the cesium based on scale model tests. Thus, DOE has committed to a 99.9% removal of the cesium in the treated supernatant and has confirmed this through operations. This satisfies the NRC recommendation expressed in the previous STS SER (Ref. 4). In addition, this cesium removal fraction helps to demonstrate decontamination of the supernatant consistent with the ALARA principle.

## 3.2 ADMINISTRATIVE CONTROLS

The general WVNS policies and procedures have been previously reviewed by NRC staff and were determined to provide a satisfactory system for protecting public health and safety (Ref. 16).

The addition of the SMWS means that the STS operators will have to be trained in the installation and operation of the sludge mobilization pumps and the caustic addition system. The operational plans specify that the SMWS operation be performed prior to STS operation with an extended period of sludge settling, supernatant sampling, and chemical analyses between the two operations. The STS and SMWS will not operate concurrently, so that process sampling is achieved prior to processing the sludge wash solutions.

The NRC staff has reviewed the outline of the STS/SMWS operator training and qualification courses and standards, as detailed in Section D.10.3.2 of the SAR. This material includes subjects that are related to public health and safety. The operator training and qualification programs discussed in the SAR identify the extent of the training and the verification of learning, retention, and skills development. The content outlines the qualification standard for both the Operator "A" and Operator "B" classifications. Since the STS/SMWS will be manned by only "A-qualified" personnel, the highest standards for both operations knowledge and skill level are provided. The program includes continuous training on new material and requalifying after one- and two-year periods. As a preventive measure, operations procedures have been adopted to ensure that intentional override of nuisance false alarms will not occur, although this is not stated in the SAR.

The pre-operational testing program includes tests of individual components plus a process operations system test. All mechanical and electrical components newly installed, including piping and wiring, will be tested to ensure correct installation. The sludge mobilization pumps have been extensively tested to verify that equipment and safety interlocks are functional. Satisfactory completion of these tests provides confidence in the ability of the system to protect public health and safety.

Normal operating procedures and emergency plans will be incorporated in accordance with the WVNS Policy and Procedures Manual (Ref. 7), which provides for controls to protect the public health and safety. Quality assurance for the WVDP is described in Volume I of the SAR (Ref. 8, Section A.12.0), and NRC independently audits processes and activities to verify that controls are appropriate.

## 3.3 ABNORMAL AND ACCIDENT CONDITIONS

In the SAR, systems or components which process, control or confine radioactivity are analyzed by quantifying a factor of safety. Radiation doses to the public and to workers on-site are estimated based on events postulated in the SAR. The SAR describes two categories of conditions for which potential radiation exposure has been evaluated for circumstances other than normal operations. The following subsections describe both categories that were evaluated: abnormal operations and accidents.

Abnormal operations are described as events during which a system or component which has a radiation safety role malfunctions or is operated incorrectly (operator error). One example is overpressurization of a zeolite (ion exchange) column. Criticality is treated separately in detail in the SAR, and is considered a special case of abnormal operations. Accidents result from the effects of natural disasters, such as a tornado, on the facility. Meteorological, historical, and seismological data are taken into consideration to determine whether or not each event is considered credible, and such data are quantified to the extent possible. A safety factor is calculated for each postulated event by comparing the magnitude of event which could result in excessive radiation doses with the magnitude of the design basis event. Rationales for choice of design basis events are described separately in Volume I of the SAR (Ref. 8).

The following discussions of performance are divided into two categories, externally caused releases and internal phenomena. Externally caused releases are those caused by natural phenomena outside the realm of operations of the STS and SMWS systems, while internal phenomena are those initiated by the operation of the process or its equipment and which may result in releases greater than those expected for normal operations.

### 3.3.1 Externally Caused Releases

The SAR identifies certain credible natural phenomena which are determined to deserve further investigation to ascertain effects on radioactive releases, and a rationale (supported by historical meteorological or seismological data) is given for the judgment of credibility. The effects of those credible natural phenomena on structures, systems and components important to safety are then determined so that a safety factor can be calculated. If the magnitude of the predicted phenomenon is less than that for which a release is determined to be possible, the design is considered safe by the SAR.

The following paragraphs discuss the effects of natural phenomena. The deleterious effects on structures, systems and components important to safety (abnormal operations) as a result of expected natural phenomena are discussed. Also discussed are the effects of those phenomena which are more properly considered natural disasters whose probability of occurrence is small (accidents).

The previous SER (Ref. 4) did not address abnormal operations which might result from natural phenomena other than natural disasters such as tornados, high winds, and seismic disturbances. In the following discussion, other phenomena are also incorporated to provide for completeness.

The credible natural phenomena which were considered in the SAR include seismic disturbances, tornados, and high winds. The potential for flooding was not considered credible in the SAR, and an acceptable rationale was posed. However, as previously stated, although flooding is not expected, the water-saturated clay around the tanks required consideration of buoyancy forces on the vaults of the tanks. The effect of consideration of buoyancy forces did not result in any detrimental effects on radiation safety.

**Lightning.** Not discussed in the SAR is the potential for lightning strikes, which could result in mechanical and/or electrical damage to exposed components or systems. (Secondary effects such as fire are discussed in the SAR and do not require further discussion.) The SAR should demonstrate some consideration of the phenomenon of lightning strikes, because of their common occurrence and the potential resultant damage.

A discussion on lightning strikes appears in supporting documentation for the sludge mobilization removal system (Reference 17, "Design Criteria Sludge Mobilization Waste Removal System"). In Section D.2.1.2, "Site Characteristics Affecting the Safety Analysis," the SAR mentions that ". . . other site-specific loads (e.g., high winds or snow loading) were also explicitly included in the design criteria, but were typically subsumed by other, more controlling loads and their associated margins of safety." It appears that lightning strikes were similarly included in the design criteria but subsumed by more controlling loads. The SAR should include a rationale for not considering the phenomenon of lightning, similar to the rationale for not considering the effect of a tornado on the control room, discussed elsewhere in the SAR.

For some components, the possibility of damaging lightning strikes is not significant because it is negated by design (such is the case for Tank 8D-1, which is underground). However, the control room and the sludge mobilization pump access appear potentially exposed to lightning strikes, however.

Any lightning strikes in the vicinity of the facility are likely to be diverted to the grounded stack, which would act as a lightning rod to ground the electrical discharge and protect equipment and personnel. This feature of protection of nearby structures, systems, and components renders the possibility of lightning damage unlikely. The effect of a lightning strike, however improbable, would be such that the process would be safely shut down until repairs or replacement could be effected. The general feature of fail-safe design would simply shut down the process until necessary repairs could be made. The fail-safe design feature is required in the design of the sludge mobilization waste removal system design criteria (Ref. 17, Section 4.9.3).

Fire protection systems which rely on electrical power for operation, alarm, and control would be particularly vulnerable to fires initiated by lightning, since the secondary effect of such a lightning strike or multiple lightning strikes might be coincidental loss of primary and emergency power while fire initiated by a lightning strike could spread. Factors which act to negate this possibility or mitigate its effects, such as backup systems and procedures, are important. These factors include (1) minimization of structures, systems and components which are combustible or susceptible to fire damage; (2) procedures to ensure that operations could safely be shut down prior to onset of an electrical storm; (3) disabling of structures, systems or components due to fire would cause the process to be safely shut down until repairs or replacement could be effected; and (4) the effects of a fire coincident with a lightning strike or multiple lightning strikes could be overridden by manual operation of control valves in the protected valve aisle beneath the control room. These factors are adequate to ensure radiological safety in the event of a fire.

**Tornados.** In addition to the control room, the only other component of the system which appears potentially exposed to tornados is the sludge mobilization pump access. In the SAR, the last sentence of Section D.9.3.5, "STS and SMWS Barrier Integrity Analysis," states that the most vulnerable link appears to be in the flexible bellows connection, yet there is no previous or further discussion. A review of Table D.9.3-1 on page 273 of the SAR shows tear of the expansion bellows at >0.5 to 1.5 times the Design Basis Tornado (DBT), which indicates a potential safety factor less than one. The SAR should include mention of the sludge mobilization pump access in its discussion on tornado missiles, along with a rationale similar to that provided elsewhere in the SAR for not considering the effect of a tornado on the control room. Due to the below-grade location of the pump access, the probability of damage due to a tornado missile is very unlikely. However, if it were to occur, the negative tank pressure induced by ventilation would act as a backup safety feature to prevent release from the tank until temporary measures could be enacted to allow repair or replacement. An explanatory note would reduce concern over the appearance of the safety factor figures in Table D.9.3-1 of the SAR. If necessary, additional ventilation capability could be brought to bear, as it is during pump installation. If the pump becomes disabled, a replacement pump or the shield plug could be re-installed in its place. Alternatively, the disabled pump could be left in place and an additional pump could be installed at another existing riser location (see Figure 6). The fail-safe nature of the process design is also a mitigating factor to any potentially deleterious effects, since waste processing systems are designed to safely stop the process until action can be taken.

**Earthquakes.** In Table D.9.3-1 of the SAR, the third heading is given as "Failure Mode of Barrier." However, the entry in the third row in the table, "Impact of the pump column at the top of the tank," will not result in failure. This is substantiated in the SAR by the text in the third paragraph of Section D.4.1.1.8. There it is noted that, although impact may occur between the Tank 8D-2 riser and the M-1 pump column at earthquake motions near 0.5 times the Design Basis Earthquake (DBE), failure will not occur in either the riser or the pump column up to 4 times the DBE. Repeated in the last paragraph of Section D.5.2.1 in the SAR is the statement that failure is not expected until 4 times the DBE. The SAR should more clearly indicate that failure is not expected by eliminating the entry in Table D.9.3-1.

**Control of Flow Direction into Tank 8D-2.** Credible accident conditions are not expected to create a pathway for leakage from the tanks (Ref. 13). The associated safety factor based on the magnitude of event is greater than 2 times the DBE, per SAR Table D.9.3-1, p. 250. As a contingency, water saturation by controlled pumping into the clay surrounding Tank 8D-2 is intended to prevent seepage from Tank 8D-2. In Section D.2.1.2, "Site Characteristics Affecting the Safety Analysis," the SAR states that "accidental liquid releases of radioactive material to the environment were determined not to be credible for the SMWS as determined by confinement barrier integrity review." This subject also appears in SAR Section D.5.3.2.4, concerning "Leak Detection Systems." Water injected around the outside of the vaults to maintain a piezometric potential is intended to cause any leakage to be from the outside to the inside.

**Safety Classification of Structures, Systems and Components.** The design, construction, and operation of WVDP structures, systems, and components is reviewed most stringently for those whose failure could result in the greatest radioactive exposure due to a credible accident (Ref. 8, Sec. A.4.4.1, p. 148). The degree of review is governed by the safety classification.

In addition to identifying those natural phenomena which were considered credible and which could result in release of radiation or radioactive exposure, the SAR separately classifies structures, systems and components with respect to the level of effective radiation dose equivalent which could happen as a result of failure. There are four classifications identified in the SAR:

- Class A: Off-site dose equivalent > 25 rem (.25 Sv)
- Class B: Off-site dose equivalent > 0.5 rem (5 mSv)
- Class C: On-site effective dose equivalent > 3 rem (.03 Sv)
- Class N: Not important to radiological safety

In comparison to dose equivalent values given in 10 CFR Part 20, the level corresponding to Class C is conservative. In 10 CFR 20.1201(a)(1)(i) (Ref. 3), the occupational dose limit for production or utilization facilities is given as 5 rem/yr (0.05 Sv/yr) for the total effective dose equivalent.

None of the structures, systems or components is relegated to Classes A or B, based on the assessment that ". . . the dose to the maximally exposed off-site individual as the result of any credible accident considered for the STS or SMWS does not exceed 5 mSv (500 mrem) annual effective dose equivalent (AEDE) . . ." (Ref. 2, p. 39). However, the classification definitions given in the SAR do not consider the credibility of failure, only the off-site dose equivalent which could result from such a failure. The definitions for each classification could be expanded to qualify their association with only credible events, since the procedure detailed in Volume I of the SAR (Figure A.4.4-1) indicates that probabilities are taken into consideration.

### 3.3.2 Internal Phenomena

A review of the STS/SMWS process was performed to determine credible internally initiated events that could allow radioactive contamination into the environment beyond what is expected through normal operation. Four event types were reviewed and are reported in the following subsections.

#### *3.3.2.1 Criticality*

The SAR considers the criticality potential in the major vessels and components with sufficient volumes for such an event. These include the Tanks 8D-1 and 8D-2, the zeolite columns, prefilter and reservoir, and postfilter. The suspension of the sludge in Tank 8D-2 during the mobilization and washing operation was analyzed. It was determined that Tank 8D-2 was safe with ten times the anticipated amount of plutonium and uranium in the tank. The material within Tank 8D-2 is considered to be safe with respect to criticality for normal and abnormal operations.

The ion exchange material is zeolite-based (Ionsiv IE-96). It appears that it will be used in all four columns, unless by sampling it is found that the Pu level in the wash solution is too high and the Pu concentration in the resultant LLW would be unacceptable. If the wash solution contains an unacceptably high concentration of Pu, these can be removed with special titanium-treated zeolite which will, apparently in this case, be used in one or more of the four zeolite columns. The incorporation of a titanium-treated zeolite to retain the Pu is a new aspect of the criticality issue. A double contingency principle for criticality control within the zeolite columns is incorporated in the operational safety requirements (OSR) program (Ref. 18). The zeolite column is the most restrictive vessel for fissile material concentration. OSR-IRTS-10 has been implemented to determine and limit the accumulation of fissile Pu to 750 g. The volume of material processed through a column will be limited, based on the measured concentration of fissile plutonium in samples taken from Tank 8D-2 before operations begin.

The distribution of Pu-loaded zeolite material in Tank 8D-1 is accomplished by use of the zeolite mobilization pumps. This process has been demonstrated to effectively distribute the spent zeolite material within Tank 8D-1. In Section D.9.2.4 of the SAR, "Nuclear Criticality," it is stated that extensive calculations indicate that the limiting mass of $^{239}$Pu inside a titanium-coated zeolite column is approximately 1.0 kg if $k_{eff} + 2\sigma$ is not to exceed 0.95. The limitation of fissile Pu accumulation to 750 g is more conservative and makes criticality unlikely.

#### *3.3.2.2 Explosions*

The potential for explosion is considered in the SAR. Specifically, the potential for explosion exists as a result of generation and accumulation of hydrogen gases resulting from corrosion and radiolytic decomposition of the water in the supernatant. Gas samples collected from 8D-2 have consistently been less than 1% hydrogen. The air movement provided by the WTFVS and the PVS effectively prevents the buildup of hydrogen gases. Hydrogen concentrations within the ventilation systems off-gas lines are monitored during operations to indicate if hydrogen is being generated within the system. The development of an explosive mixture and its ignition are not considered to be credible under normal or abnormal operations.

#### *3.3.2.3 Fires*

The STS/SMWS facilities are constructed with limited amounts of combustible material, and administrative procedures control the introduction and accumulation of combustible material in the STS/SMWS structures. There is a potential that maintenance operations which involve welding, cutting, or grinding could provide an ignition source. The procedures incorporated for such maintenance activities assure that appropriate fire prevention measures are taken.

The fire detection equipment, alarm systems, and suppression systems, as discussed in the SAR, provide appropriate monitoring and response measures to limit the amount of damage a fire could do if one should ignite. The most severe credible accident of a fire in a WTFVS HEPA filter would result in a 0.5 mrem (5 $\mu$Sv) committed effective dose equivalent (CEDE) at the nearest residence.

The ventilation system incorporates fire dampers which automatically operate to isolate areas and prevent the spreading of a fire. Since the ventilation air flow is from a nonradioactive area into a radioactive area, the isolation of a fire will limit the amount of radiation protection that is lost by the event which causes areas to be isolated for ventilation. The continued testing and training of fire prevention and fire fighting techniques will assure that fires are not likely to occur and become a concern to public health and safety.

#### 3.3.2.4 *Over-Pressurization of Vessels and Piping*

The estimated release of 1% of the zeolite if the column is over-pressurized is a conservative estimate. The possibility of over-pressurization is extremely remote. The over-pressurization event, should it occur, is treated as an explosion with release through the ventilation system, for which the off-site dose is estimated to be extremely small (Ref. 19).

Abnormal operations with respect to the sparge line are discussed in SAR Section D.9.1.8. The fail-safe design concept in this case pre-empts the potential for over pressurization of the zeolite column. This is made possible by maintaining the STS utility air pressure (100 psig) less than the STS process piping design pressure (150 psig) and much less than the zeolite column design pressure (250 psig).

#### 3.3.2.5 *SMWS Pumps — Vibration and Hydraulic Forces*

A comprehensive vibration analysis of the pump and pump shaft has been performed, and critical pump speeds have been locked out of the control system so that pump speed increases with a step function through such critical speeds.

The possibility of blockage or restriction of flow from one of the two pump nozzles has been considered in the SAR. In Section D.4.1.1.6 "Mobilization Pumps," the force generated by one nozzle having all the flow discharging from it (i.e., one of the two nozzles plugged) is given as 468 pounds. This force is discussed in "8D-2 Sludge Mobilization Confinement Barrier Integrity Review" (Ref. 13), on page B-10, where its source is attributed to the pump manufacturer. However, there is an apparent inconsistency in Reference 13 in the unbalanced pump load assumed in the design of the pump support structure given in Section B.2.2. The text of that section states that hydrodynamic loads due to unbalanced forces were "estimated at 20% of the assumed jet force of 1600 pounds in designing the pumped [sic] shaft and pump support structure." Also, the table summarizing the pump loads shows the hydrodynamic unbalanced horizontal force at the pump to be 16 kips. It appears that the entry in the table should be 1.6 kips, to be consistent with the text of the same section. Although the 1600 pound figure is conservative when analyzing the support structure, it is not conservative when analyzing the deflection of the 50-foot long pump column to determine whether or not it would impact a tank-bottom fin. It is evident that the force generated by one nozzle having all the flow discharging from it (i.e., one of the two nozzles plugged) should be 468 pounds. The "assumed" value in Section B.2.2 on Page B-8 of the Appendix of Reference 13 should be consistent with the 468-pound value to avoid the implication that the pump shaft will deflect and impact with fins at the bottom of the tank.

### 3.4 ALARA REVIEW

The concept of ALARA is a philosophical principle of radiation protection. As such, there are no set "ALARA Level" standards to use for the review of this SAR. Because of the subjective nature of an ALARA review, most of the comments in this section are to confirm that a comprehensive ALARA program is in effect at the WVDP in general, and regarding the STS and SMWS in particular. The main objective with all these comments is to interpret the design and the operation of the STS and the SMWS and determine whether the levels achieved here were truly "as low as reasonably achievable."

An effective ALARA program requires a number of actions. A record of radiation doses must be kept, and the implementor must show measures used to track and, if necessary, reduce exposures to radiation. Both the magnitudes of average and maximum individual doses should be recorded. It should be demonstrated that periodic reviews of performance have been made, with the effort to achieve ALARA. An example for which an effective ALARA program should be applied is the step-wise installation of the SMWS Pumps (refer to Section 3.1, page 13).

As stated previously in Section 3.1 of this SER, the pre-operational test acceptance criteria for the ion exchange columns include an extraction efficiency of 99.9% of the cesium in the treated supernatant and has confirmed this through operations. This proven cesium removal fraction helps to demonstrate decontamination of the supernatant consistent with the ALARA principle.

DOE may consider a numerical cost-benefit analysis or an optimization analysis, which may include trade-off studies comparing various options, for an effective ALARA program.

In the SAR, Section D.8.3, "Radiation Protection Design Features," discusses the ALARA plan for both the STS and the SMWS. The WVDP ALARA program is sufficient, however, DOE may consider adding an operational trend analysis to their ALARA program. The operational trend analysis as described in 10 CFR Part 20 (Ref. 3) is not strictly required for the WVDP, in accordance with the provisions of the Memorandum of Understanding between NRC and DOE (Ref. 20). Adoption of operational trend analysis offers many advantages, such as the following:

- Demonstrates a commitment to assure the best protection of radiological health and safety of the public;
- Demonstrates a commitment to assure the best achievable radiological health and safety of the employees;
- Ensures efficient overall project management by integration, analysis, and improvement of work practices and environment.

In section D.8.2.2, the SAR has indicated that the airborne radioactivity levels are less than 0.1 times the derived air concentration (DAC) for normal and abnormal operations. The DAC has been discussed in 10 CFR Part 20 (Ref. 3). The levels of airborne radioactivity described in section D.8.2.2 of the SAR are within limits established by the NRC in 10 CFR Part 20 (Ref. 3).

In the SAR, Section D.8.6.3, "Estimated Exposures from Airborne Releases," provides estimated exposures from airborne releases and also refers to SAR Table D.8.6-1 for the maximum normal operations air releases of the major nuclides. Values given in Table D.8.6-1 reflect the effluent release during the total period of operations (18 months), which DOE may consider indicating in the Table.

Subpart D Section 20.1302 of 10 CFR Part 20 (Ref. 3) provides a compliance guideline for dose limits for individual members of the public. This Subpart requires not just the final figures but also a demonstration that all factors have been accounted for. The SAR provides an acceptable maximum individual off-site effective dose limit for all radionuclides to be 70 $\mu$rem (0.7 $\mu$Sv) annually. This value was calculated by AIRDOS-PC (Ref. 21). In order to verify the acceptability of all the values listed in Table D.8.6-1 of the SAR, a comparison of the data with the values given in Table 2, Column 1, of 10 CFR Part 20 (Ref. 3) was done after requesting the values from DOE in units of Ci/ml. The calculation of the annual dose limits are affected by many factors, such as the physical and chemical characteristics of the effluent (e.g. aerosol size distribution, solubility, density, radioactive decay equilibrium, chemical form, etc.). Additional factors include predominant wind direction and velocity, rain fall pattern and rate, radius of controlled area or restricted area (the "closest" member of public). It was determined that these calculated values are well within the regulatory values.

## 3.5 CONCLUSIONS OF THE SAFETY REVIEW

A comprehensive review of the SAR has been performed to evaluate the modifications to the STS by the installation of the SMWS. The main modifications include (1) the addition of caustic solution to Tank 8D-2 to control and limit Pu solubility; (2) the installation of sludge mobilization pumps to agitate the solid wastes and wash them of their sodium sulfate salts; and (3) the possible employment of titanium-coated zeolite to remove soluble Pu from the wash solution. This SER presents the results of the comprehensive review, summarized in the following paragraphs. The review concludes that protection of public radiological health and safety is not reduced by these modifications.

Examination of the gaseous effluents from the sludge washing and wash treatment operations reveals that the site boundary dose is far below allowable limits. The calculated maximum individual off-site effective dose equivalent for all radionuclides is less than 70 $\mu$rem (0.7 $\mu$Sv) annually.

Examination of the potential for and effects of abnormal and accident conditions from both externally and internally caused events reveals that site boundary dose is small. The most severe credible accident of a fire in a WTFVS HEPA filter would result in a 0.5 mrem (5 $\mu$Sv) committed effective dose equivalent (CEDE) at the nearest residence.

The experience gained through operation of the STS since 1988 for the removal of cesium from the supernatant can be applied to the continued operation of the system for removal of the sodium sulfate salts. The solubility of the plutonium and strontium in the wash solution is an important parameter for control. The use of titanium-coated zeolite to remove the soluble transuranics in order to meet NRC's requirements for stable low-level radioactive waste form disposal is contingent on the levels of soluble Pu measured in the wash solution. The NRC expects DOE to accurately measure, record, and report the plutonium levels in the wash solution and STS treated water, and to ascertain when its removal becomes necessary to prevent transport to the LWTS.

The caustic addition to the supernatant in Tank 8D-2 will limit the solubility of Pu and reduce its transport from the HLW. The effect of the caustic on the potential for corrosion potential of Tank 8D-2 (due to caustic stress corrosion cracking) is considered to be minimal. However, activities to monitor the three types of corrosion are encouraged.

The sludge mobilization pumps have been thoroughly designed, manufactured, and tested. The pumps are adequate for agitation of the sludge. The process for their installation is complicated by the requirement to lower them in several steps into the solid sludge. This process can be monitored by the WVDP's ALARA program to effectively control radiation exposure to site workers.

The employment of titanium-coated zeolite is necessary only if Pu solubility cannot be limited by pH control of the wash solution. This contingency has been adequately accounted for in the various site administrative controls and operating procedures. Additional operator and supervision training will be required and can be performed under the existing WVDP Training Programs and Requirements, which are discussed in the SAR Section D.10.

The review of accident conditions that are potentially threatening to STS equipment and operation suggests that lightning strikes also be considered, since they are a credible natural phenomenon. The effect of a tornado missile on the SMWS pump access and enclosure is also suggested for consideration since this is a possible event.

Finally, the review recommends that operational trend analysis be added to the WVDP's ALARA program, described in Reference 3. The adoption of operational trend analysis can enhance worker and public radiological health and safety.

# 4 DECONTAMINATION AND DECOMMISSIONING

The STS/SMWS will be used to remove the radioactive wastes from the storage tanks and then will have to be decontaminated for decommissioning. These later activities will also have the potential for impacting public health and safety. The WVDP Act requires that the NRC prescribe the decontamination and decommissioning requirements to the DOE. Several parts of NRC regulations call for decommissioning plans, however there are no specific requirements for the content of these plans. The NRC is currently investigating the rules that would establish criteria for decontamination and decommissioning of nuclear facilities. Three criteria that were developed and used for the previous SER (Ref. 4) were examined again. The criteria are:

(1) Did DOE take measures during the design and construction of the STS/SMWS to facilitate eventual decontamination and decommissioning of the facility;

(2) Does DOE plan to take measures during the operation of these systems to facilitate eventual decontamination and decommissioning of the facility; and

(3) Has DOE prepared a preliminary decontamination and decommissioning plan?

Many measures have been taken during the design phases for the STS and the SMWS that will allow for decontamination. The use of stainless steel components, epoxy paint or liners, and mechanical shaft seals will limit the amount of contamination during operation. The STS design criteria, WVNS-DC-013, Rev. 1, Sec. 2.3.5.4, para. D, (Ref. 22), require that the zeolite column design ". . . must not prevent addition of an elution process (the elution hardware will not be designed into the column at this time)." This indicates that the materials of construction of the column (as well as other components and plumbing of the STS and SMWS) must be such that they will be compatible with possible materials which will be used to decontaminate them.

Other design features, such as remote maintenance capability, reduced crud traps or sludge collection pockets, and installed flushing capability, have been shown to be able to reduce the amount of decontamination that will be required upon decommissioning. The potential for accumulation of HLW in the sludge mobilization pump shaft enclosure has been reduced through the use of a mechanical seal and flushing lines. Maintenance of a shaft seal during operation of the pumps will be required to control the spread of contamination.

The West Valley Demonstration Act directs that NRC provide guidance to DOE to develop the decontamination and decommissioning requirements for WVDP. As operations continue, it will be necessary for DOE to have operating procedures that maintain records of contamination levels and drawings of as-built configurations.

The WVDP Long-term Radioactive Waste Management Plan (Ref. 12) has been revised several times to incorporate new requirements and controls. This plan should be revised, as necessary, as new decontamination and decommissioning criteria are established by the NRC.

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# 7 NOMENCLATURE — LIST OF ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AEDE | Annual Effective Dose Equivalent |
| ALARA | As Low As Reasonably Achievable |
| CEDE | Committed Effective Dose Equivalent |
| Cs | Cesium |
| $Cs^{137}$ | Isotope of Cs with atomic mass = 137 |
| CFR | Code of Federal Regulations |
| CSS | Cement Solidification System |
| DAC | Derived Air Concentration |
| DBE | Design Basis Earthquake |
| DBT | Design Basis Tornado |
| DC | Drum Cell |
| DOE | Department of Energy |
| g | Gram |
| HEPA | High Efficiency Particulate Air |
| HLW | High-Level Radioactive Waste |
| IRTS | Integrated Radwaste Treatment System |
| $k_{eff}$ | Multiplication Factor |
| kg | Kilogram |
| Kips | Kilopounds |
| LWTS | Liquid Waste Treatment System |
| LLW | Low-Level Radioactive Waste |
| m | Meter |
| ml | Milliliter |
| mrem | Millirem |
| NRC | Nuclear Regulatory Commission |
| OSR | Operational Safety Requirements |
| PUREX | Plutonium Resin Extraction |
| PVS | Permanent Ventilation System |
| Pu | Plutonium |
| $Pu^{139}$ | Isotope of Pu with atomic mass = 139 |
| SAR | Safety Analysis Report |
| SCFM | Standard Cubic Feet per Minute |
| SER | Safety Evaluation Report |
| SMWS | Sludge Mobilization and Wash System |
| Sr | Strontium |
| STS | Supernatant Treatment System |
| THOREX | Thorium Resin Extraction |
| TRU | Transuranic |
| VP | Vitrification Process |
| WC | Water Column |
| WTF | Waste Tank Farm |
| WTFVS | Waste Tank Farm Ventilation System |
| WVDP | West Valley Demonstration Project |

## 7 NOMENCLATURE — LIST OF ACRONYMS AND ABBREVIATIONS (cont'd)

| | |
|---|---|
| WVNS | West Valley Nuclear Services |
| $\mu$rem | Microrem |
| $\mu$Sv | Microsievert |
| $\sigma$ | Standard Deviation, as used in Statistical Calculations |