DEC 0 3 2004

Mr. Robert A. Pedde, President
Westinghouse Savannah River Company
Aiken, South Carolina 29808

Dear Mr. Pedde:

SUBJECT: Submittal of the 2004 Annual Update of Saltstone Safety Basis Documents and USQE Summary Report (Letter, French to Hansen, CBU-WSD-2004-00027, 11/9/2004)

The Department of Energy Savannah River Operations Office approves the Saltstone Facility Documented Safety Analysis (WSRC-SA-2003-00001) Revision 2 based on the enclosed Safety Evaluation Report. This revision shall be implemented within 30 days.

The action directed herein is considered to be within the scope of work of the existing contract. If the Contractor considers that carrying out this direction will increase contract costs or delay any delivery, the Contractor shall promptly notify me orally, confirming and explaining the notification in writing as soon as possible, but within no more than five (5) working days. Following oral notification and submission of the written notice of impacts, the Contractor shall await further direction from me.

If you have any questions, please contact me or have your staff contact Hope Franklin at 208-0972.

Sincerely,

Original Signed by
Charles E. Anderson
for Jeffrey M. Allison
Manager

WDED:MHF:kl

WDED-05-12

Enclosure:
Saltstone Safety Evaluation
Report, Revision 1, Supplement 1

cc w/o Encl:
W. J. Johnson, WSRC, 730-1B
H. T. Conner, Jr., WSRC, 730-1B
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WDED Rdg File
AMWDP Rdg File
MGR's Rdg File
DMC, DOE-SR, 730-B
ECATS

Safety Evaluation Report
Revision 1, Supplement 1

For the Savannah River Site
Saltstone Facility
Operated by WSRC under Contract No. DE-AC09-96SR18500

Documented Safety Analysis
WSRC-SA-2003-00001, Revision 2

November 2004

Prepared by: Melissa H. Franklin for

Melissa H. Franklin

Reviewed by: Tom Temple

Tom Temple

Approved by:

Michael A. Mikolanis, Director
Engineering Division

THE OFFICE OF THE ASSISTANT MANAGER FOR WASTE DISPOSITION PROJECT
SAVANNAH RIVER OPERATIONS OFFICE
U.S. DEPARTMENT OF ENERGY

EXECUTIVE SUMMARY

This Safety Evaluation Report (SER) supplement documents the U.S. Department of Energy (DOE) evaluation of the safety basis (SB) document annual update submitted in Reference 1. The annual update incorporates new source term concentrations in support of the Low Level Waste (LLW) Transfer from H-Canyon to Tank 50 and Low Curie Salt (LCS) processing, allowing receipt of low level waste with greater concentrations of chemicals and radionuclides than were previously evaluated and incorporates new consequences from the resulting revised accident analysis. This SER was prepared in accordance with Savannah River Implementing Procedure (SRIP) 400, Chapter 421.1, "Nuclear Safety Oversight" (Reference 2).

The scope of the evaluation focused on the changes made to the Saltstone safety basis to document acceptance of low level waste with higher source term concentrations. Accordingly, this SER serves as a supplement to the Saltstone Facility SER, Revision 1 (Reference 3) and documents the basis for approval of the submitted changes.

No Conditions of Approval were identified as a result of this review.

DOE has reviewed and determined that no change to the Saltstone Technical Safety Requirements (S-TSR-Z-00002, Revision 1) is required as a result of the submitted change package.

Final approval of the SB document change package and this SER supplement by the DOE-SR Manager is in accordance with Savannah River Manual 300.1.1B, Chapter 1, "Functions, Responsibilities, and Authorities Procedure" (Reference 4).

1.0 INTRODUCTION

This SER supplement documents the DOE evaluation of the annual update to the Saltstone Facility Documented Safety Analysis (DSA) (WSRC-SA-2003-00001, Revision 2) for acceptance of low level waste with higher source term concentrations to support the LLW Transfer from H-Canyon to Tank 50 and LCS processing. The higher source term concentrations were derived from H-Canyon LLW and LCS tank data, the concentrations are intended to bound LLW and LCS streams that will be processed at the Saltstone Facility (Reference 5).

This annual update submittal was delayed from August to November as requested by WSRC in Reference 6. The request for delay was approved by DOE in Reference 7. The delay in submittal was needed to allow receipt of sample analysis results to ensure the increased source term concentrations would bound not only constituents of LCS but also the LLW Transfers from H-Canyon.

This annual update does not address the Saltstone Potential Inadequacy in the Safety Analysis/Discovery Unreviewed Safety Question (USQ), USQ-SSF-2004-0020. The Saltstone Facility has identified the potential for benzene generation and accumulation that could result in an explosion in the saltstone vaults. The current Saltstone DSA does not postulate an explosion in the vaults.

Compensatory measures have been put in place and analysis is in progress to resolve the Discovery USQ. This issue will be addressed in a future DSA revision.

This SER supplement documents the basis for approval of the submitted safety basis changes to allow receipt of low level waste with higher source term concentrations.

2.0 REVIEW CRITERIA AND SUMMARY OF CHANGES

The annual update package was reviewed to ensure compliance with appropriate DOE criteria: 10CFR830, DOE Guide 421.1-2 (Reference 8), Saltstone DSA requirements, and technical accuracy. In addition, the annual update package was reviewed for consistency, completeness, adequacy of justification, documentation, and reasonableness.

The documentation submitted in Reference 1 for DOE review included the following changes:

1) Hazard and accident analysis revised to reflect source term and consequences resulting from higher bounding chemical and radionuclide concentrations.
2) Deleted paragraph describing the Vault 1 roof from the facility description.
3) Incorporations of revisions as part of the 2004 annual update (e.g. editorial corrections, organizational updates, etc.)

The annual Saltstone Facility Unreviewed Safety Question Evaluation (USQE) Summary Report was submitted along with the DSA annual update. DOE reviewed a sampling of the USQEs, no issues were identified.

3.0 EVALUATION OF DOCUMENT CONTENT AND CONCLUSIONS

The Saltstone Facility revised the bounding source term concentrations to allow receipt of low level waste with greater concentrations of chemicals and radionuclides. The concentrations chosen to support processing LLW from H-Canyon and LCS material that may be sent to the Saltstone Facility were based on H-Canyon LLW and LCS tank data with a factor of conservatism. The hazard and accident analysis consequences were revised to reflect the higher source term.

DOE reviewed the Consolidated Hazard Analysis (CHA), WSRC-TR-2001-00574, Rev. 4, and supporting Consolidated Hazard Analysis Process (CHAP) Basis calculation, S-CLC-Z-00035, Rev. 0. The CHA and the CHAP Basis calculation were revised to update the hazard analysis for increased chemical and radiological source term. No new accident scenarios were introduced and the Salt Feed Tank (SFT) Explosion remains the bounding accident scenario. The bounding radionuclide and chemical concentration tables in the DSA were revised to reflect the higher source term.

The bounding consequences associated with an atmospheric release of a single liter of feed material were calculated using the revised Saltstone radiological inventory in calculation S-CLC-Z-00028, Rev. 1. The source term and radiological consequences resulting from an explosion in the SFT were calculated

in S-CLC-Z-00033, Rev. 1. This revision to S-CLC-Z-00033 corrected mathematical errors made in the previous revision in computing the source term resulting from resuspension. Source term from a benzene explosion is 15 gallons, subsequent resuspension is assumed to occur for a period of 8 hours resulting in a source term of 0.25 gallons. The total resulting source term is 15.25 gallons. In reviewing S-CLC-Z-00033, DOE noted that an incorrect dose factor was used in calculating the Total Effective Dose Equivalent (TEDE) to the offsite receptor due to resupsension. However, the TEDE to the offsite receptor from resuspension is so small (10^{-6} mrem) it is negligible to the total dose to the offsite receptor. This dose factor error will be corrected in the next revision of the calculation. The dose factors from S-CLC-Z-00028 and the source term from S-CLC-Z-00033 were used to derive the resulting radiological consequences at 30m, 100m, and offsite. The revised Saltstone radiological inventory and the mathematical error correction resulted in an increase in consequence shown below in Table 1. The increase in consequence does not challenge the evaluation guidelines.

Table 1: Bounding Radiological Consequences

Receptor Location	Radiological Consequences DSA Rev. 1	Radiological Consequences DSA Rev. 2
30m	6 rem	8 rem
100m	0.6 rem	0.8 rem
Offsite	1.3 mrem	1.8 mrem

DOE review of calculation S-CLC-Z-00033 found that the source term from the postulated explosion was conservatively derived by assuming the SFT was completely filled with a stoichiometric concentration of benzene (maximized explosion energy), assumed the waste to be vaporized had the density of water (versus salt solution), and used the TNT equivalent methodology consistent with DOE-HDBK-3010. Additionally, the resuspension portion of the source term conservatively assumed the spilled contents evaporated for 8 hours and used the bounding Airborne Release Rate from DOE-HDBK-3010. For both sources, the Release Fraction and Leak Path Factor were assumed to be 1.0. DOE review of calculation S-CLC-Z-00033 found that the receptor dose derived from the released source term used appropriate factors for breathing rate, surface roughness, release time, and used the conservative dose conversion factors for 1 micron particle sizes for both the offsite and onsite receptors. The offsite atmospheric dispersion used the 95th percentile meteorology consistent with DOE-STD-3009, Appendix A. The onsite atmospheric dispersion used 50th percentile meteorology consistent with SRS site practices as described in Reference 9 [letter, Shingler to Hansen, 11/11/2004, FSS-2000-00018]. Given the conservatisms in the source term and receptor dose calculations, and the fact that the 100m dose was determined to be less than 1 rem, the use of the 50th percentile meteorology is judged acceptable.

In the bounding accident, an explosion in the SFT, it is assumed that 15.25 gallons of respirable salt solution could be released to the environment. The SFT can contain up to 6504 gallons of salt solution. The beyond Design Basis Accident (DBA) was revised to reflect the increased source term concentrations. The beyond DBA scenario atomizes the entire 6504 gallons of salt solution in the

SFT, which results in an increase in public dose from 564 mrem to 781 mrem. This beyond DBA event does not challenge the evaluation guideline limit of 25 rem to the public.

Based on the radiological inventory and projected worst case accident consequences, Saltstone is classified as a Hazard Category 3 (HC3) nuclear facility per DOE-STD-1027-92. By definition HC3 facilities do not contain sufficient fissile materials to present a criticality hazard. Inventory limits specified in Chapter 3 of the DSA bound the amount of fissile materials introduced into the facility. The Nuclear Criticality Safety Evaluation (NCSE) described in Chapter 3 concludes that a criticality is not a credible accident. DOE reviewed the Saltstone NCSE (N-NCS-Z-0001, Rev. 2) and concluded it was consistent with the requirements of DOE-STD-3007.

The chemical consequences associated with an explosion in the SFT were calculated in S-CLC-Z-00034, Rev. 1 to reflect the revised chemical inventory to determine if onsite or offsite guidelines were exceeded. Previously all organic chemicals in solution were assumed to be volatile. This revision changed the treatment of three chemicals that were previously assumed volatile chemicals to non-volatile chemicals. Tetraphenylborate, tributylphosphate, and EDTA have no vapor pressure, therefore, they are treated as non-volatile. Based on the calculation, DOE concludes Saltstone does not contain sufficient chemical concentrations to challenge onsite or offsite limits.

No changes were needed to the Technical Safety Requirements (TSR). The Saltstone Facility has only administrative controls. The Waste Acceptance Program administrative control ensures the composition of waste streams received into the facility is within analyzed radionuclide and chemical concentration limits specified in the DSA. The implementing document of this control, the Saltstone Waste Acceptance Criteria (WAC) will be revised to protect the higher concentrations analyzed by the DSA and allowed by the Saltstone environmental permit. Tank 50 is periodically sampled to ensure WAC compliance.

The Vault 1 roof was deleted from the facility description section of the DSA. The weather protection roof was removed during the 0.4 LCS project to permit installation of a roof suitable for use with 0.4 curie per gallon grout. This project was cancelled due to the high cost associated with replacing the Vault 1 roof and new vault construction. Removal of the weather protection roof was completed, but a replacement roof was not installed. Before grout can be disposed of in Vault 1 a weather protection or permanent roof will need to be installed. This change to the DSA is purely a descriptive change, the weather protection roof is not credited with any safety function.

Other minor changes (e.g. editorial corrections, organizational updates, etc.) were also included as a part of the annual update. DOE reviewed these changes and found them acceptable; however, no specific discussion here is warranted.

4.0 DOE COMMENT RESOLUTION AND DOCUMENT STATUS

As a result of the review, DOE did not identify any comments or outstanding issues requiring a revision to the submitted documents.

5.0 CONDITIONS FOR APPROVAL

No Conditions for Approval were identified as a result of this evaluation.

6.0 CONCLUSION

Based on the above evaluation, DOE concludes Revision 2 of the Saltstone Facility DSA, WSRC-SA-2003-00001, is acceptable. The increase in radionuclide and chemical concentration does not introduce any new accidents or challenge any evaluation guidelines. The CHA, DSA, and supporting calculations have been appropriately updated to reflect the change in radionuclide and chemical concentration. Saltstone Facility operations can be conducted without undue risk to the offsite population, the onsite worker, or the environment.

7.0 REFERENCES

1. Letter, French to Hansen, "Submittal of the 2004 Annual Update of Saltstone Safety Basis Documents and USQE Summary," CBU-WSD-2004-00027, November 9, 2004.

2. Savannah River Implementing Procedure, "Nuclear Safety Oversight," SRIP 400, Chapter 421.1, Revision 3, October 3, 2003.

3. Letter, Allison to Pedde, "Submittal of Revision 1 of the Documented Safety Analysis (DSA) and Technical Safety Requirements (TSR) for the Saltstone Facility (Letter, French to Hansen, CBU-WSD-2003-00034, 8/26/03)," DC-03-015, August 27, 2003.

4. Savannah River Manual 300.1.1B, Chapter 1, "SR Functions, Responsibilities, and Authorities Procedure," August 6, 2004.

5. Memo, Chandler to Thompson, "Recommended Saltstone Facility Radiological and Chemical WAC and Permit Limits for Development of Tank 50H Material Balance," WSP-SSF-2004-00015, July 15, 2004.

6. Letter, French to Hansen, "Delay of Annual Update of Saltstone Safety Basis Documents and USQE Summary Report - 2004," CBU-WSD-2004-00023, August 24, 2004.

7. Letter, Hansen to French, "Delay of Annual Update of Saltstone Safety Basis Documents and USQE Summary Report – 2004 (Your Letter, CBU-WSD-2004-00023, 8/24/04)," DC-04-055, September 3, 2004.

8. DOE Guide 421.1-2, "Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10CFR830," October 24, 2001.

9. Letter, Shingler to Hansen, "Basis for 50 Percent Meteorology in the Savannah River Site (SRS) Safety Basis Calculations for Worker Analyses," FSS-2000-000018, November 11, 2004.

AUG 2 7 2003

Mr. R. A. Pedde, President
Westinghouse Savannah River Company
Aiken, SC 29808

Dear Mr. Pedde:

SUBJECT: Submittal of Revision 1 of the Documented Safety Analysis (DSA) and Technical Safety Requirements (TSR) for the Saltstone Facility (Letter, French to Hansen, CBU-WSD-2003-00034, 8/26/03)

The Department of Energy approves the Saltstone Facility DSA (WSRC-SA-2003-00001) Revision 1, and TSR (S-TSR-Z-00002) Revision 1 based on the enclosed Safety Evaluation Report. These revisions shall be implemented within 30 days.

The action taken herein is considered to be within the scope of the existing contract and does not authorize the Contractor to incur any additional costs (either direct or indirect) or delay delivery to the Government. If the Contractor considers that carrying out this action will increase contract costs or delay any delivery, the Contractor shall promptly notify the Contracting Officer orally, confirming and explaining the notification in writing within five (5) working days. Following submission of the written notice of impacts, the Contractor shall await further direction from the Contracting Officer.

If you have any questions, please call me or your staff may call Dave Faubert at 208-0140.

Sincerely,

Original signed by
Charles E. Anderson
for Jeffrey M. Allison
Manager

WDED:DRF:kl

DC-03-015

Enclosure:
SER

cc w/o encl:
H. T. Conner, Jr. WSRC, 730-1B
W. J. Johnson, WSRC, 703-H
L. J. Hollick, WSRC, 730-1B
W. S. Elkins, WSRC, 730-1B
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bcc w/o encl:
D. R. Faubert, WDED
WDED Rdg File
AMWDP Rdg File
Mngr Rdg File
DMC + DMB Rdg File
ECAT

of Revision 1 of the Documented Safety Analysis (DSA) and Technical Safety Requirements (TSR) for the Saltstone Facility," Dated" AUG 27 2003

Safety Evaluation Report
Revision 1

for the
Savannah River Site

Saltstone Facility
Operated by Westinghouse Savannah River Company
Under Contract No. DE-AC09-96SR18500

Documented Safety Analysis
WSRC-SA-2003-00001
Revision 1

Technical Safety Requirements
S-TSR-Z-00002
Revision 1

August 2003

Prepared by: ______________________
David R. Faubert

Reviewed by: TC Temple
Tom C. Temple

Approved by: ______________________
Michael A. Mikolanis, Director
Waste Disposition Engineering Division

THE OFFICE OF THE ASSISTANT MANAGER FOR WASTE DISPOSITION PROJECT
SAVANNAH RIVER OPERATIONS OFFICE
U.S. DEPARTMENT OF ENERGY

Executive Summary

This Safety Evaluation Report (SER) revision documents the basis for the U.S. Department of Energy (DOE) approval of the Saltstone Facility (Saltstone) Documented Safety Analysis (DSA) WSRC-SA-2003-00001, Revision 1 and associated Technical Safety Requirements (TSRs) S-TSR-Z-00002, Revision 1. These revisions were prepared to allow processing of Low Curie Salt solution. These revisions (1) reduced the material at risk from 60,000 gallons of salt solution to 15,000 gallons (principally by removing the Salt Solution Hold Tank from the process flow sheet), (2) added an Inadvertent Transfer from Tank 50H administrative control program, and (3) changed the bounding accident from a full facility fire (less than 10 mrem Maximally Exposed Offsite Individual (MOI)) to an explosion in the Salt Feed Tank (1.3 mrem MOI).

The purpose of the Saltstone DSA is to describe the design and safety analysis of the facility in sufficient detail to demonstrate the facility has been constructed and can be operated, maintained, shut down, and decommissioned safely and in compliance with applicable laws and regulations. The DSA also derives the management and administrative controls necessary to ensure the safe operation of Saltstone. The Saltstone DSA and TSRs were prepared by Westinghouse Savannah River Company (WSRC), the primary contractor for management and operation of the Savannah River Site (SRS) located near Aiken, South Carolina.

The Saltstone Facility is part of the overall High Level Waste System. Saltstone receives liquid waste from Tank 50 and produces a non-hazardous grout. Tank 50 contains wastes from the abandoned In-Tank Precipitation project, receives waste from the Effluent Treatment Facility (ETF), and will receive waste feed from low curie salt and actinide removal processes in the future. The Saltstone Facility includes both the processing facility and the disposal vaults which permanently store the grout waste form.

The Saltstone DSA/TSRs evaluated and described in this SER constitute the 10 CFR 830 compliant safety basis documents for Saltstone. Based on the radioactive material inventory of the facility and projected worst case accident consequences, Saltstone is classified as a Hazard Category 3 nuclear facility in accordance with the guidelines of DOE-STD-1027-92. Due to the limited inventory of nuclear materials, the consequence of accident analysis is limited to local consequences only. No accident scenario approaches the limits needed for Safety Class or Safety Significant Structures, Systems, or Components (SSCs). The safety documentation is built around the inventory control program which limits the facility inventory to that of a Hazard Category 3 Nuclear Facility. Given the limited hazards, a DSA commensurate with facility hazards was developed.

The Saltstone Facility TSRs are limited to administrative controls for facility inventory and site wide programs. The accident analysis did not result in the need for Safety Limits, Limited Conditions for Operations or Surveillance Requirements.

As a result of the DOE review, DOE approves the Saltstone DSA and TSR as the safety basis documents. The basis for DOE approval is that the accident analysis is complete and the controls are commensurate with the hazards. All major programmatic elements covered in the DSA are adequate to support safe operation of the Saltstone Facility. In addition, DOE review of the TSRs concluded that the administrative controls have been properly identified and contain the proper linkage to the accident analysis documented in the DSA.

I. Background and Review Process

A. Introduction

Revision 0 of the Saltstone DSA and TSRs were approved by DOE letter PC-03-035, dated July 7, 2003. Subsequently, facility modifications and re-analyses were prepared to allow processing of Low Curie Salt solution. The DSA and TSR revisions to reflect this (1) reduced the material at risk from 60,000 gallons of salt solution to 15,000 gallons (principally by removing the Salt Solution Hold Tank from the process flow sheet), (2) added an Inadvertent Transfer from Tank 50H administrative control program, and (3) changed the bounding accident from a full facility fire (less than 10 mrem Maximally Exposed Offsite Individual (MOI)) to an explosion in the Salt Feed Tank (1.3 mrem MOI).

This SER revision (Revision 1) documents the basis for DOE approval of: (1) the Saltstone Facility Documented Safety Analysis (DSA), WSRC-SA-2003-00001, Revision 1, August 2003, and (2) the Saltstone Facility TSRs, S-TSR-Z-00002, Revision 1, August 2003. These Saltstone DSA and TSR revisions were initially submitted to DOE for approval on July 31, 2003 in letter CBU-WSD-2003-00029. DOE comments on this initial submittal and WSRC responses are included as Appendix A of this SER. The documents were resubmitted to DOE on August 26, 2003 in letter CBU-WSD-2003-00034 to incorporate the resolution of these comments.

The purpose of the Saltstone DSA is to describe the design and safety analysis of the facility in sufficient detail to demonstrate the facility has been constructed and can be operated, maintained, shut down, and decommissioned safely and in compliance with applicable laws and regulations. The DSA also defines the management and administrative controls necessary to ensure the safe operation of the Saltstone Facility. The Saltstone DSA/TSRs evaluated and described in this SER constitute the Nuclear Safety Rule 10 CFR 830 compliant DSA/TSR.

B. Facility Description

The Saltstone Production and Disposal Facility is located in Z-Area on the Savannah River Site. The facility consists of a processing facility (210-Z), and disposal facility (vaults). Storage Silos exist for the dry feed material used in grout production along with a Salt Feed Tank (SFT) which receives liquid waste transfers (through an inter-area line) from Tank 50. The Salt Solution and dry materials are mixed together in the enclosed processing facility and transferred by pumps to the disposal vault. Saltstone processes waste fed through Tank 50 from the Effluent Treatment Facility and legacy waste from the In-Tank Precipitation process. Future plans include the processing of Low Curie Salt and Actinide Removal Process filtrate.

C. Document Content and Conclusion

The Saltstone DSA describes the facility hazards and accident scenarios considered and qualitatively analyzed. None of the postulated events resulted in a challenge to the evaluation guidelines for the public or co-located worker. One event, the explosion of the SFT was considered to have the potential to challenge the evaluation guidelines for the facility worker. This event was further analyzed and it was determined to be less than the evaluation guidelines. The resulting consequences of this event were determined to be 1.3 mrem to the public and 0.6 rem to the co-located worker and 6.0 rem to the facility worker.

Based on the radiological inventory and projected worst case accident consequences, Saltstone is classified as a Hazard Category 3 nuclear facility per DOE-STD-1027-92.

The Saltstone DSA concludes that the hazard and accident analyses demonstrate that the administrative controls and safety programs are in place to protect onsite workers, offsite public, and the environment from radiological and chemical hazards such that operation of Saltstone presents an acceptable level of risk. The DSA is formatted in accordance with DOE-STD-3009-94.

The Saltstone TSRs set forth the administrative controls (ACs) that are used by facility operators to ensure the inventory is maintained below Hazard Category 3 limits. Saltstone ACs protect the health and safety of the public and onsite workers from undue exposure to radiological and chemical hazards. The TSRs ACs are formatted in accordance with DOE Guide G 423.1-1.

D. DOE Review Criteria

The DOE review of the Saltstone DSA was performed against the criteria contained in DOE-STD-3009-94. Review of the Saltstone TSRs was performed against the criteria contained in DOE Guide G 423.1-1. The intent of this TSR review was to confirm the DOE Guide G 423.1-1 requirements were met in terms of content and organization, and that the TSRs were consistent with the derived controls in the DSA. The specific criteria from DOE-STD-3009-94 and DOE Guide G 423.1-1 utilized for this review are identified in SER Section II for each DSA Chapter and TSR section.

This SER was prepared by the Savannah River Operations Office (SR) in accordance with guidance from DOE-STD-1104-96 and SRIP 400, Chapter 421.1. In accordance with SRM 300.1.1A, the SR Manager is the approval authority for the DSA and TSRs.

E. DOE Conditions of Approval

As a result of the DOE review, no conditions of approval were determined.

F. Summary of DOE Evaluation

DOE approval is based on the determination that the hazards and accident analysis is complete commensurate with the hazards of the facility. The derived administrative controls limit the hazards associated with the facility. Defense-in-depth concepts are in place where appropriate and all other major programmatic elements covered in the DSA have been deemed adequate to support safe operation of Saltstone. In addition, DOE review of the TSRs concluded that the administrative controls have been properly identified and developed and contain the proper linkage to the accident analysis documented in the DSA. The risk associated with the potential hazards of operating the Saltstone Facility is acceptable.

II. Approval Basis

The five approval bases required by DOE-STD-1104-96 are described in Sections II.A through II.E below.

A. Base Information (DSA Chapters ES, 1, and 2)

The DSA contains sufficient background and fundamental information to support the review of the technical aspects contained in Chapters 3, 4, 5, and the TSRs. Most of the base information is contained in the Executive Summary (ES), Chapter 1 (Site Characteristics), and Chapter 2 (Facility Description). These chapters were reviewed against the DOE-STD-3009-94 criteria and the results have been summarized below.

Executive Summary

Evaluation

The Executive Summary meets the guidance given in DOE-STD-3009-94 by giving appropriate background information and describing the facility's mission in disposing of low level radioactive liquid waste from the HLW system. An overview of the facility is given along with an adequate discussion of the segmentation (2 segments) and associated hazard category. The Saltstone Facility is a Hazard Category 3 Nuclear Facility in accordance with DOE-STD-1027-92. The safety analysis overview summarizes the primary hazards of the facility and identifies the administrative and defense in depth controls established to prevent/mitigate event occurrence. Due to the limited hazards, no Safety Class or Safety Significant SSCs are required. The worst case credible accident of a SFT explosion is identified in this section and is described in Chapter 3 in more detail. The Executive Summary concludes that the Saltstone Facility can be operated without undue risk to the public or onsite workers.

The DSA is organized into six chapters. Chapters 1 – 5 follow the format and scope outlined in DOE-STD-3009-94 with Chapter 6 being a combination of what would normally be Chapters 6-17. Due to the hazard category of the facility and straightforwardness of the process, use of the graded approach is supported.

DOE has determined that the Executive Summary meets the guidance of DOE-STD-3009-94.

Chapter 1, Site Characteristics

Evaluation

The Site Characteristics chapter gives adequate detail about the location of the facility relative to the site boundary and neighboring facilities. Details such as demography, meteorology, and geology, while not required for this facility due to localized consequences only, are available in the referenced Generic Safety Analysis Report section. Natural and man-made accident event initiators are identified that were considered as part of the accident analysis, Chapter 3.

DOE has determined that Chapter 1 meets the guidance of DOE-STD-3009-94.

Chapter 2, Facility Description

Evaluation

Chapter 2 follows the format and content given in DOE-STD-3009-94 by providing a facility overview and a description of the structures within Z-Area and major components. A description of the individual processes with the facility is discussed along with diagrams of the facility and process. The chapter covers the process from dry material receipt and storage through the processing facility to the disposal vaults. Adequate detail is given to complete an assessment of normal operations for which safety programs will be devised. Adequate discussion is given for the facility confinement system, safety support systems, utility distribution system, and major system components.

DOE has determined that Chapter 2 meets the guidance of DOE-STD-3009-94.

B. Hazard and Accident Analysis (DSA Chapter 3)

The purpose of this DSA chapter is to provide information that will satisfy the requirements of 10 CFR 830 to evaluate normal, abnormal, and accident conditions, including consideration of: natural and man-made external events; identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials; and consideration of the need for analysis of accidents which may be beyond the design basis of the facility. This chapter describes the process used to systematically identify and assess hazards to evaluate the potential internal, man-made external, and natural events that can cause the identified hazards to develop into accidents. This chapter also presents the results of this hazard identification and assessment process. Hazard analysis considers the complete spectrum of accidents that may occur due to facility operations; analyzes potential accident consequences to the public and workers; estimates likelihood of occurrence; identifies and assesses associated preventive and mitigative features; identifies safety significant SSCs; and identifies a selected subset of accidents, designated accidents, to be formally carried over into the accident analysis. Subsequent accident analysis evaluates these accidents for comparison with the Evaluation Guidelines (EG). This chapter covers the topics of hazard identification, facility hazard categorization, hazard evaluation, and accident analysis.

B.1 Criteria: DOE-STD-3009-94, para. 3.1, Introduction, and 3.2, Requirements

This section provides an introduction to the contents of this chapter and includes the objectives and scope specific to the chapter as developed. This section lists the design codes, standards, regulations, and DOE Orders that are required for establishing the safety basis of the facility. Standards/Requirements Identification Document (S/RIDS) may be referenced as appropriate.

Evaluation

The introduction and requirements section of Chapter 3 discuss the process used to identify and assess the hazards associated with the facility. Hazard identification, categorization, hazard evaluation, and accident analysis is discussed. Appropriate requirements are listed as the basis for this evaluation: 10CFR830, STD-3009-94, STD-1027-92, STD-5502-94, and S/RIDs. DOE concludes that the criteria are adequately addressed.

B.2 Criteria: DOE-STD-3009-94, para. 3.3.1, Methodology

This section identifies the method used by analysts to identify and inventory hazardous materials and energy sources in terms of quantity, form, and location associated with facility processes or associated operations. The method used to screen out standard industrial and insignificant hazards is presented.

Evaluation

The methodology used (Hazard and Operability Analysis) for the hazard evaluation was based on DOE-STD-3009-94, the WSRC 11Q manual, and the Consolidated Hazards Analysis Process Methodology Manual. The following criteria was used to determine that hazards are common industrial hazards or routinely accepted hazards (must meet at least one):

1. The hazard is routinely encountered first-hand by the general public
2. Public consensus standards exist to control the hazard
3. No evidence exists that there are public or employee concerns about the hazard.
4. The hazard is subject to OSHA regulations.

DOE reviewed the Consolidated Hazards Analysis and compared it to Chapter 3. Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.3 Criteria: DOE-STD-3009-94, para. 3.3.2.1, Hazard Identification, and 3.3.2.2, Hazards Categorization

A summary table identifying hazards in terms of quantity, form, and location is to be provided. The basic set of radionuclides, hazardous chemicals and flammable and explosive materials used or potentially generated in facility processes should be identified, and any mechanical, chemical, or electrical source of energy that may influence accident progression involving such materials are included. The facility hazards classification, and where segmentation has been employed, the segment boundaries and individual segment classification are included and is justified.

Evaluation

Two tables are given for the bounding radiological and chemical concentrations. These concentrations are controlled via the Saltstone Waste Acceptance Criteria (WAC) which the sending facility (i.e., Tank 50) must comply with. A bounding quantity of 15,000 gallons of salt solution is allowed in the facility. DOE judges this quantity to be conservative because it is twice the maximum volume of the SFT, and additional hold up in the processing area is minimal. The assumed radionuclide concentrations were found to be consistent with the values in Chapter 3. Other key inputs for tank volume and temperature were found to be consistent with or conservative to actual conditions. Criticality is not a concern at the facility based on the low fissile concentration and the lack of concentrating mechanisms. This conclusion is supported by the Saltstone NCSE (N-NCS-Z-0001).

The Saltstone facility is broken down into two segments for hazard category analysis: the Saltstone Production Facility (i.e., processing area) and the Saltstone Disposal Facility (i.e., the disposal vaults). The boundary between the segments is defined where the grout free-falls from the transfer line into the vault. Saltstone is correctly categorized as a Hazard Category 3 Nonreactor Nuclear facility. From a chemical prospective, Saltstone chemical quantities were compared to 29CFR1910, 40CFR68, 40CFR302, and 40CFR355. Based on this analysis Saltstone was categorized as a Low Hazard Chemical Facility (EM-STD-5502).

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.4 Criteria: DOE-STD-3009-94, para. 3.3.2.3.1, Planned Design & Operational Safety Improvements

Planned improvements not yet implemented are identified and the basis for committing to the improvement and, if needed, any interim controls proposed until the improvement is implemented, is summarized.

Evaluation

Selected vault cells are being upgraded to allow processing of low curie salt waste. This modification includes the addition of roughing filters on the cell vents and installation of a cell drainage system.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.5 Criteria: DOE-STD-3009-94, para. 3.3.2.3.2, Defense in Depth

Significant aspects of defense-in-depth are summarized, and associated safety-significant SSCs and other items needing TSR coverage are identified and distinguished from SSCs contributing to Defense in Depth (DID). Facility design and administrative features of defense-in-depth are included.

Evaluation

Based on the results of the hazards analysis, Saltstone does not have any Safety Class or Safety Significant SSCs. The facility does have several DID SSCs and administrative programs identified in the DSA; however, no credit was taken for these features. The Hazards Analysis tables included in Chapter 3 of the DSA describe the DID SSCs and administrative programs.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.6 Criteria: DOE-STD-3009-94, para. 3.3.2.3.3, Worker Safety

Major features protecting workers from the hazards of facility operation, exclusive of standard industrial hazards, are summarized and administrative features in terms of the programmatic elements covered in later chapters of the DSA are categorized.

Evaluation

The TSR administrative controls (Saltstone Waste Acceptance Criteria and Inadvertent Transfers controls) limit the radionuclides and chemical inventory in the facility. These controls, in addition to the Safety Management Programs covered in Chapter 6, protect workers from the hazards of facility operations.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.7 Criteria: DOE-STD-3009-94, para. 3.3.2.3.4, Environmental Protection

Pathways for uncontrolled release to the environment are documented and potential consequences and preventive and mitigative features associated with those pathways are qualitatively estimated.

Evaluation

This section of the DSA discusses preventative and mitigative features for the release of offgases from the Saltstone process. The Process Vessel Ventilation System receives offgases from the SFT and exhaust from the Saltstone Hold Tank Ventilation System, which are sent through a HEPA filter. Liquid process waste from flushing and facility drains are collected and sent to the SFT. Rainwater is collected in sumps and analyzed for chemical and radiological contamination. This rainwater can be sent to the SFT. Leaching and migrations from the Saltstone vaults are minimized through the vault design and grout composition. Groundwater monitoring wells are used to detect contamination.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.8 Criteria: DOE-STD-3009-94, para. 3.3.2.3.5, Accident Selection

Accidents to be further evaluated are to be identified and the process for selecting these accidents should be described.

Evaluation

No postulated events challenged the evaluation guidelines for the public or co-located worker. No quantitative analysis is required. One event (Explosion in the SFT) was of concern to the facility worker and controls for this event have been previously evaluated (Section B.6).

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.9 Criteria: DOE-STD-3009-94, para. 3.4.1, Methodology

Computer codes used to quantify the consequences of operational accidents, natural phenomena, and external events are identified and described. Methodology used to estimate radiological or other hazardous material source terms for DBAs is documented, and includes: 1) the basic approach for estimating physical facility damage from DBAs; 2) the general basis for assigning Material-At-Risk (MAR) quantities; and 3) the basis for material release and respirable fractions or release rates used. Methods used to estimate dose and exposure profiles include meteorological conditions, time dependent characteristics, activity, and release rates or duration for radioactive or other hazardous materials that could be released to the environment and are documented.

Evaluation

Detailed accident analysis is not required for Hazard Category 3 facilities; however, the DSA references the Consolidated Hazards Analysis (CHA) for details of accidents and events qualitatively evaluated. For the worst case accident for the facility, which is an explosion in the SFT, the MACCS computer code version 1.5.11.1 was utilized to determine the resulting consequence. The methodology employed is consistent with analysis guidance in DOE-STD-3009-94 and DOE-HDBK-3010.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.10 Criteria: DOE-STD-3009-94, para. 3.4.2, Design Basis Accidents (DBA)

Each DBA, including natural phenomena hazards, is identified and facility and equipment response (emphasizing preventative or mitigative equipment) to the event is summarized. All parameters and phenomenological models used to derive the source term are defined; exposures and doses are derived and compared to the EGs; and safety class SSC and assumptions judged to require TSR coverage are identified.

Evaluation

No events at Saltstone challenge the EG's so a full analysis of facility events is not required. The worst case accident is an explosion in the SFT, which bounds all other events. This accident is below EG's for requiring safety class or safety significant SSCs.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

B.11 Criteria: DOE-STD-3009-94, para. 3.4.3, Beyond DBAs

Evaluate accidents beyond DBA to provide a perspective of the residual risk associated with the operation of the facility.

Evaluation

In the bounding accident, explosion in the SFT, it is assumed that 15 gallons of salt solution could be released to the environment. The SFT can contain up to 6504 gallons of salt solution. The beyond DBA scenario atomizes the entire 6504 gallons of salt solution in the SFT, which results in a public dose of 564 mrem. This beyond DBA event does not challenge the evaluation guideline limit of 25 rem to the public.

Based on the information presented, DOE concludes that the criteria are adequately addressed.

C. Safety Systems, Structures, Components (DSA Chapter 4)

The purpose of this chapter is to provide details on those facility structures, systems, and components that are necessary for the facility to satisfy EGs, provide defense in depth, or contribute to worker safety. Descriptions are provided of the functional requirements and performance criteria required to support the safety functions identified in the hazard and accident analyses and to support subsequent derivation of TSRs.

Evaluation

The hazards at Saltstone are limited and below the severity needed to require Safety Class or Safety Significant SSCs. DOE has reviewed the hazards and concurs with this conclusion.

D. Derivation of Technical Safety Requirements (DSA Chapter 5)

D.1 Purpose

The purpose of this DSA chapter is to provide information necessary to adequately describe the derivation of the Technical Safety Requirements (TSRs). The information satisfies the requirements of 10 CFR 830, "Nuclear Safety Management", Subpart B Section 830.205. 10 CFR 830 requirements are amplified in Appendix A to Subpart B, Section G and Table 4, and further specified in DOE-STD-3009-94, Chapter 5.0.

This chapter builds upon the control functions determined to be essential in Chapter 3, Hazard and Accident Analysis" and Chapter 4, "Safety Structures Systems, and Components," to derive TSRs. This chapter supports and provides the necessary information for the determination of TSRs which consists of summaries and references to pertinent sections of the Documented Safety Analysis (DSA) in which design (SSCs) and administrative features (non-SSCs) are required to prevent or mitigate the consequences of accidents. Design and administrative features addressed include ones which: (1) provide significant defense-in-depth; (2) provide for significant worker safety; or (3) provide for the protection of the public. Expected products of this chapter (based on a graded approach) include:

- Information with sufficient basis from which to derive any of the TSR parameters for Safety Limits, Limiting Control Settings, Limiting Conditions for Operation, and Surveillance Requirements.
- Information with sufficient basis from which to derive TSR administrative controls or to specify programs necessary to perform institutional safety functions.
- Identification of passive design features addressed in the DSA.
- Identification of TSRs from other facilities that affect the facility's safety basis.

10 CFR 830 Subpart B and Appendix A to Subpart B specify that the safety analysis thoroughly explore the safety acceptability of all modes of operation, set points and operational parameters, combinations of inoperable equipment, staffing and qualification levels of operating crews, and limitations of administrative controls to verify that operation anywhere within the envelope will afford adequate safety provisions. Safety analyses should furnish the information necessary to validate, confirm, derive or modify the bases for TSRs.

The DOE conducted a thorough in-depth review of the Saltstone Facility DSA Chapter 5 to ensure the credited controls identified in Chapter 3 were properly used to form the basis for the TSRs to documents compliance with the DSA requirements of 10 CFR 830 for derivation of TSRs.

D.2 <u>Acceptance Criteria and Evaluation</u>

As stated above, the DSA must meet the requirements from DOE-STD-3009-94, Chapter 5.0, for the derivation of TSRs.

D.2.1 <u>Introduction (DOE-STD-3009-94, Section 5.1)</u>

<u>Criteria</u>

This section shall provide an introduction to the contents of this chapter based on the graded approach and includes objectives and scope specific to the chapter as developed.

<u>Evaluation</u>

Section 5.1 describes the purpose of Chapter 5 to document the derivation of TSRs and provide a link between the DSA and TSRs. The DOE review found this section adequately meet the requirements of DOE-STD-3009-94, Section 5.1.

D.2.2 Requirements (DOE-STD-3009-94, Section 5.2)

Criteria

This section shall list the design codes, standards, regulations, and DOE Orders that are required for establishing the safety basis of the facility. The intent is to provide only the requirements that are specific for this chapter and pertinent to the safety analysis, and not a comprehensive listing of all-industrial standards or codes or criteria. S/RIDs may be referenced as appropriate.

Evaluation

The DOE reviewed section 5.2 of the chapter and found it adequately meets the requirements of DOE-STD-3009-94, section 5.2. Review of Chapter 5 noted the referenced Regulation (10CFR830), DOE Standard 3009-94, and DOE Guide 423.1-1 to be appropriate for this chapter. No other requirements are pertinent to the safety analysis.

D.2.3 TSR Coverage (DOE-STD-3009-94, Section 5.3)

Criteria

This section shall provide assurances that TSR coverage for the facility is complete. This section lists the features identified in Chapters 3 and 4 of the DSA that are needed to provide significant defense in depth, provide for significant worker safety, and provide for significant public safety. Associated TSR SLs, LCSs, LCOs, Surveillance Requirements, Administrative Controls and Design Features are to be included in this presentation. This section will specifically note those safety SSCs listed, if any, that will not be provided with TSR coverage and provide accompanying explanation.

Evaluation

Section 5.3 identified that through the Saltstone Facility CHA no accident scenarios were identified that would challenge the onsite or offsite EGs. Therefore, it was concluded that no SC or SS SSCs were needed and two Administrative Controls involving the Saltstone WAC Program and Inadvertent Transfers from Tank 50 controls were identified. DOE reviewed the CHA and reached the same conclusion. Section 5.3 as written adequately meets the requirements of DOE-STD-3009-94, section 5.3.

D.2.4 Derivation of Facility Modes (DOE-STD-3009-94, Section 5.4)

Criteria

This section shall derive basic operational modes (e.g., startup, operation, shutdown) used by the facility that are relevant to derivation of TSRs. The definition of modes required in this subsection expands and formalizes the information provided in Chapter 3, "Hazards and Accident Analyses", regarding operational conditions associated with accidents.

Evaluation

This section provides adequate justification for not developing TSR Modes. Operating modes are not required to implement the Administrative Controls (ACs) in the TSR and modes to differentiate between processing and not processing will not improve safety. The DOE review concluded that section 5.4 adequately meets the requirements of DOE-STD-3009-94.

D.2.5 TSR Derivation (DOE-STD-3009-94, Section 5.5, 5.5.X, 5.5.X.1, 5.5.X.2, 5.5.X.3)

The information can be organized by hazard protected against, specific features, or by TSRs.

- Applicable Hazards/Features/TSR "X"
- Safety Limits (SLs), Limiting Control Settings (LCSs), Limiting Conditions of Operation (LCOs), and Surveillance Requirements (SRs)
- Administrative Controls.

Criteria (a) - Applicable Hazards/Features/TSR (5.5.X)

This subsection identifies the specific feature(s) listed from DOE-STD-3009-94, Section 5.3 and the relevant modes of operation.

Criteria (b) - SLs, LCSs, LCOs, and SRs (5.5.X.1 and 5.5.X.2)

This section shall provide the basis and information sufficient to derive Safety Limits, Limiting Control Settings, Limiting Condition for Operations, and Surveillance Requirements to support the facility TSR document required by 10 CFR 830. Safety analyses should furnish the information necessary to validate, confirm, derive or modify the bases for TSRs.

Evaluation for Criteria (a) and (b)

The section concludes that since no offsite EGs are challenged the only controls needed are for protection of the Facility Worker. This protection will be accomplished through the WAC program and Inadvertent Transfers from Tank 50 controls as part of the Administrative Control programs. In addition, since no SC or SS SSCs were identified, no LCOs were identified. The DOE review found this section to adequately meet the requirements of DOE-STD-3009-94.

Criteria (c) - Administrative Controls (5.5.1)

This section provides the basis and identifies information necessary to derive TSR Administrative Controls. This section is the only applicable section for those features listed in section 5.3, "TSR Coverage," that are provided with only TSR administrative controls. The rationale necessary for assigning TSR Administrative Controls needs to be clearly and briefly stated. 10 CFR 830, Subpart B, Appendix A, Table 4, identifies the necessary information to include in the Administrative Control section.

The Administrative Control section of the TSR document will contain commitments to establish, maintain, and implement these programs at the facility and, as appropriate, facility staffing requirements.

Evaluation

The DOE reviewed DSA Chapter 3 as well as the needed safety programs/controls to confirm that section 5.5.1 adequately captured the needed Administrative Controls and that the control was properly included in the Administrative Control section of the Technical Safety Requirements. The DOE review found this section to adequately meet the requirements of DOE-STD-3009-94 for content.

D.2.6 Design Features (DOE-STD-3009-94, Section 5.6)

Criteria

This section shall identify and briefly describe the passive design features that, if altered or modified, would have a significant effect on safe operation. Simply reference Chapter 2, "Facility Description" if that chapter contains the desired information.

Evaluation

No Design Features were identified as stated in this section. The DOE review found DSA section 5.6 to adequately meet the requirements of DOE-STD-3009-94 for the content of this information.

D.2.7 Interface With TSRs from Other Facilities (DOE-STD-3009-94, Section 5.7)

Criteria

This section shall summarize TSRs from other facilities that affect this facility's safety basis and briefly summarize the provisions of those TSRs.

Evaluation

This section discusses Saltstone's interface with Tank 50, as well as the specific controls. Concentration, Storage, and Transfer Facilities (CSTF) TSRs (S-TSR-G-00001) Administrative Controls require the contents of Tank 50 to meet the Saltstone WAC prior to transferring to Saltstone. The DOE review found this section to adequately meet the requirements of DOE-STD-3009-94, section 5.7.

Conclusion

Chapter 5 of the DSA is acceptable. Based on the DOE review, the purpose and required elements of Sections 5.1 and 5.2 of this SER have been satisfied.

E. Chapter 6, Safety Management Programs

E.1 Safety Management Programs

The purpose of this chapter is to provide a summary description of the key features of the various site specific, programmatic safety programs as they relate to the Saltstone Facility. This Chapter consolidates the information normally identified separately in DOE-STD-3009-94 Chapters 7-17. Chapter 6 was developed using the graded approach guidelines for a Hazardous Category 3 (HC3) facility. Although consolidated into one chapter, the DSA contains the required information identified in the DOE guidance. All sections within DSA Chapter 6 were evaluated against the applicable requirements in DOE-STD-3009-94. The evaluation of several of the key chapters are documented below.

E.2 DOE-STD-3009-94, Chapter 6, Criticality

By definition HC3 facilities do not contain sufficient fissile materials to present a criticality hazard; therefore, this section is not applicable to the Saltstone Facility. Inventory limits specified in Chapter 3 of the DSA will control the amount of fissile materials introduced into the facility. The Nuclear Criticality Safety Evaluation (NCSE) described in Chapter 3 concludes that a criticality is not a credible accident. DOE reviewed the Saltstone NCSE (N-NCS-Z-0001) and concluded it was consistent with the requirements of DOE-STD-3007.

Based upon DOE review no further information is needed to satisfy the requirements of Chapter 6.

E.3 DOE-STD-3009-94, Chapter 7, Radiation Protection Program

Sections 6.3.3-6.3.10 reference the site level WSRC Procedure Manual 5Q, for implementation of the Radiation Protection Program at the Saltstone Facility. The Saltstone Facility follows the SRS guidelines, as outlined in WSRC 5Q and other WSRC procedures, established for radiation exposure control, radiological monitoring, instrumentation, record keeping, and occupational exposures. The sitewide Radiation Protection Services is the consulting organization for implementation of the Radiation Protection Program at Saltstone and in some program areas is responsible for direct implementation of program objectives.

Radiation exposure limits are established to minimize the potential stochastic effects and to prevent deterministic effects. Saltstone Facility personnel are responsible for controlling and minimizing external radiation exposures. The Saltstone Facility follows the site administrative limits, radiological practices, dosimetry, and respiratory protection outlined in WSRC 5Q.

Penetrating radiation at the Saltstone Facility exists as gamma radiation resulting from the beta-gamma decay of radionuclides in the salt solution and saltstone grout process streams and in the solidified saltstone. Exposures to this radiation cannot be eliminated but is reduced through facility design and administrative control levels. Examples of Saltstone Facility features that limit worker exposure to radiation include: 1) the SFT was designed with increased wall thickness for shielding and 2) access to the process area containing the saltstone mixing equipment is minimized while the process is in operation. The Saltstone Facility was designed such that all continuously occupied areas have a dose rate less than 0.5 mrem/hr and that all intermittently occupied areas (10% of the workday) have a dose rate less that 5.0 mrem/hr.

Based on DOE review, the criterion to summarize the radiation program has been met.

E.4 DOE-STD-3009-94, Chapter 8, Hazardous Material Protection

Sections 6.4.1-6.4.11 reference the site level WSRC Procedure Manual 4Q for the implementation of the Hazardous Material Protection Program at the Saltstone Facility and references the Standard/Requirements Identification Document (S/RID) for the relevant codes and standards for hazardous material protection policies and programs.

The Saltstone Facility follows the SRS guidelines, as outlined in WSRC 4Q and other WSRC procedures, established for hazardous material protection program including: non-radiological ALARA, training, exposure control, monitoring, instrumentation, record keeping, and hazard communication. Industrial Hygiene (IH) programmatic functions and field activities are managed by the Environmental, Safety, and Heath Services (ES&H) Department, which is a part of the Field Support Business Unit. The site level program provides industrial hygienists for the Saltstone Facility.

Based on DOE review, the criterion to summarize the hazardous material protection program has been met.

E.5 DOE-STD-3009-94, Chapter 9, Radioactive and Hazardous Waste Management

Sections 6.5.1-6.5.2 adequately address the introduction and requirements respectively. Section 6.5.3 discusses the radioactive and hazardous waste management program and organization and also lists the strategy WSRC has for achieving SRS waste management objectives. This strategy includes: attempt to identify a method for disposing of the waste before it's generated, reduce/minimize waste generation, recycle/reuse, segregate at generation point, treat to minimize mobility and impact to environment, and dispose of in monitored repositories. Radioactive, hazardous, and mixed waste management at the Saltstone Facility is the responsibility of the Operations organization.

Facilities that send salt solution to the Saltstone Facility are required to implement a waste compliance plan (WCP) implementing the requirements of WSRC Procedure

Manual 1S. The compliance plan identifies the organization; methodologies, procedures, and training needed to implement their compliance program. The WCP ensures that the salt solution has been properly identified and characterized in compliance with the waste acceptance criteria for the Saltstone Facility.

Section 6.5.4 discuss the radioactive and hazardous waste streams and sources. The Saltstone Facility HVAC system is designed to filter regulated area air through HEPA filters before being exhausted to the stack. Chapter 2 also provides a description of the HVAC system.

In the course of Saltstone Facility operations, small quantities for radionuclides may be released to the atmosphere through the main stack at the Process Building (210-Z) and through the stack at the Operations Building (704-Z) from the laboratory fume hoods. No other gaseous hazardous, radioactive, or mixed waste effluents are generated in the Saltstone Facility.

The saltstone process does not normally generate any liquid radioactive, hazardous or mixed wastes. Based on sample results, collected rainwater may be processed along with the salt solution or discharged into the site storm drains.

The Saltstone Facility does not normally generate any solid waste. However, some maintenance activities will generate solid waste from work on contaminated systems or work on process systems. The waste is placed in approved containers (e.g., B-25 boxes) for disposal at the SRS Solid Waste Disposal Facility.

Solid hazardous waste may be generated during maintenance when rag or other absorbent materials are used to soak up solvents. The waste is drummed and shipped to an onsite SRS storage facility.

Solid mixed waste is generated in small quantities when hazardous solvents are used for radioactive decontamination. Solid mixed waste is shipped to an onsite SRS storage facility for temporary storage until final disposition is determined.

The annual average volume of solid waste generated by Saltstone Facility is forecast to be approximately 800 ft^3. All waste is characterized as required by the WSRC Procedure Manual 1S.

Based on DOE review the criterion to summarize the radioactive and hazardous waste management program has been met.

E.6 DOE-STD-3009-94, Chapter 15, Emergency Preparedness Program

Review of Emergency Preparedness Hazards Assessment (EPHA) support the information provided in Section 6.11 regarding the Saltstone Facility. The Saltstone Facility has been assessed and found to have no hazardous materials present that would require a specific Emergency Planning Zone nor an emergency plan and implementing procedures.

Based on DOE review, the criterion for the emergency preparedness program has been met.

E.7 DOE-STD-3009-94, Chapter 16, Provisions for Decontamination and Decommissioning

Planning for D&D of the Saltstone Facility will be initiated prior to termination of facility operations. The Z-area Closure Plan was submitted to DOE and SCDHEC in 1995 as an addendum to the permit application for the Saltstone Disposal Facility. Per DSA Section 6.12.3, any changes to the mission of the Saltstone Facility will be evaluated relative to design and administrative feature for facilitating D&D activities. D&D will be carried out for the existing structures for the SPF. Plans for decommissioning SPF will be developed as the facility approaches the end of its useful life. The SDF is a permanent disposal facility and will undergo an environmental closure.

Based on DOE review the criterion for the provisions decontamination and decommissioning has been met.

III Technical Safety Requirements

1. Purpose and Discussion

In accordance with 10 CFR 830 and DOE G 423.1-1, Technical Safety Requirements (TSR) shall establish safety limits, operational limits, and administrative controls necessary for the safe operation of a nuclear facility. This shall include use and application provisions, design features, related bases, and surveillance requirements necessary to protect the health and safety of the public and to minimize the potential risk to workers from the uncontrolled release of radioactive or other hazardous materials. The facility specific DSA will serve as the source document for input into the TSRs. Operation within the bounds of the TSRs will provide reasonable assurance that the nuclear facility will not threaten the health and safety of the public or pose an undue risk to workers from uncontrolled releases of radioactive or other hazardous materials.

It is important to develop the TSRs judiciously and not as a vehicle to cover the many procedural and programmatic controls inherent in the operation. In areas that the DSA does not directly supply all of the input for the TSRs, such as surveillance intervals and surveillance acceptance criteria, national and international codes, standards, and guides are to be used wherever possible. Use of a value less conservative than that expressed in applicable codes, standards, and guides should be justified in the DSA. Where conflicts exist, the selection of a particular code, standard, or guide should be justified; normally the most conservative should be selected. Where no code, standard, or guide is applicable, other documents such as risk assessments and manufacturer documentation, may serve as a basis; a justification should be placed in the DSA.

2. Acceptance Criteria and Evaluation

2.1 Use and Application (DOE G 423.1-1)

Criteria

This section of the TSR document shall contain the basic instructions for using and applying the safety restrictions contained in the TSRs, definitions of terms, operational modes, and frequency notations.

Evaluation

The DOE reviewed section 1.0 of the TSRs to ensure consistency with the guidance provided in DOE G 423.1-1 as well as to ensure the information provided was accurate and applicable to the configuration and operations of the Saltstone Facility. The section gives a brief overview of facility purpose and states that the TSRs are required to maintain the Hazard Category 3 status. The only TSR controls required are administrative. The DOE review concluded that this section adequately meets the requirements of DOE G 423.1-1.

2.2 Safety Limits (SL) (DOE G 423.1-1)

Criteria

This section should describe as precisely as possible the parameters being limited and state the limit in measurable units. This section should also include an applicability statement, which shall consist of a simple list of modes or other conditions for which the Safety Limit is applicable. Action Statements are to be included and shall completely describe the actions to be taken in the event the Safety Limit is not met. The actions should bring the affected parameter immediately within the Safety Limit and should effect a shutdown of the affected system(s) within a justified facility specific time frame. A statement prohibiting restart must be included either in the Action Statement or may be in the Administrative Controls.

Evaluation

Section 2.1 of the TSR document states that the safety analysis did not identify any single limit that met the criteria to be designated as a Safety Limit (SL); therefore no SLs are required for the Saltstone Facility. The DOE review found this section to adequately meet the requirements of DOE G 423.1-1.

2.3 Limiting Control Settings, Limiting Conditions for Operation, Surveillance Requirements, and Bases (DOE G 423.1-1)

This section shall contain the Limiting Control Settings (LCSs), and the Limiting Conditions for Operation (LCOs), as well as Surveillance Requirements.

Criteria (A) - Limiting Control Settings
LCS statements should describe, as precisely as possible, the parameter being controlled and it's limit, or the limiting setting of the device to control it.

Evaluation

As defined by DOE G 423.1-1, LCSs are associated with SLs. Since no SLs were identified for Saltstone, there are no LCSs.

Criteria (B) - Limiting Conditions for Operations

LCO statements should describe as precisely as possible, the lowest functional capability or performance level of equipment required for continued safe operation of the facility. LCO Mode Applicability Statements consist of a simple listing of the modes or conditions for which the LCO is applicable. Action Statements completely describe the action to be taken in the event that a LCO is not met.

Criteria (B.1) - Applicability

This section should contain a simple listing of the Modes or Conditions for which the LCO is applicable.

Criteria (B.2) - Actions

This section should contain ACTION statements that describe the actions to be taken in the event the LCO statement is not met.

C (B.3) - Surveillance Requirements

This section of the TSRs shall provide the Surveillance Requirements (SRs) relating to test, calibration, or inspection to ensure that the necessary operability and quality of safety related SSCs and their support systems required for safe operation of the facility. This section shall contain the requirements necessary to maintain operation of the facility within the LCOs.

Criteria (B.4) - Bases

The Bases shall provide summary statements of the reasons for the LCOs and associated surveillance requirements. The bases shall show how the numeric value, the condition, the surveillance, and ACTION statements fulfill the purpose derived from the safety documentation. The Bases shall reference the more detailed basis in the derivation of the TSRs in the DSA. The Bases shall also provide justification for the Action Times allowed when the LCO Condition statements are met.

Evaluation

The DOE reviewed sections 3.0, 4.0, and the corresponding Bases of the TSRs. The document states that, using TSR selection criteria and methodology based on 10 CFR 830, no components or parameters were identified requiring LCOs. In addition, since no LCOs were identified, no SRs are necessary. The DOE review noted this conclusion as appropriate and consistent with Chapter 3 of the DSA. Based on the DOE review, the Saltstone Facility LCOs/SRs/Bases section was found to adequately meet the requirements of DOE G 423.1-1.

2.4 Administrative Controls (DOE G 423.1-1)

Criteria

This section shall contain the administrative requirements necessary to ensure TSR compliance. Any Administrative Controls relied upon in the safety analyses shall be identified and summarized.

Evaluation

The DOE review found this section to adequately meet the requirements of DOE G 423.1-1. A few key Administrative Controls are:

- Waste Acceptance Criteria – This program ensures that the composition of waste received into the facility has been properly analyzed and is within the specified limits prior to acceptance. This will ensure the facility maintains the Hazard Category 3 designation.
- Inadvertent Transfers from Tank 50H – This programmatic control ensures inadvertent transfers are prevented, therefore protecting the assumed material at risk. This will ensure the facility maintains the Hazard Category 3 designation.

The administrative controls identified in DSA Chapter 5 as well as those specifically required by DOE G 423.1-1 (e.g., Staffing, TSR Violation processing, etc.) were verified by DOE to be appropriately included in section 5 of the TSRs. Furthermore, DOE compared the TSRs ACs to the guidance in the 5/20/03 EM-1 guidance memorandum for ACs, and found them consistent.

2.5 Design Features (DOE G 423.1-1)

Criteria

This section describes in detail those features not covered elsewhere in the TSRs that, if altered or modified, would have a significant effect on safety. This includes vital passive safety SSCs and configuration or physical arrangement.

Evaluation

Section 6.0 states no passive design features or passive SSCs were identified for inclusion in the TSRs. The DOE review found this consistent with DSA Chapter 5 Section 6.0 and adequately met the requirements of DOE G 423.1-1.

Conclusion

The TSRs were found to be adequate and consistent with DOE Guide G423.1-1. Based on the DOE review documented above, the purpose and required elements of this DSA and TSRs have been satisfied. The Saltstone Facility hazards have been appropriately evaluated and the facility can be safely operated as a Hazard Category 3 Nuclear Facility.

APPENDIX A

DOE REVIEW COMMENTS
WSRC-SA-2003-00001, Revision 1
S-TSR-Z-00002, Revision 1

Comment No.	Document/ Section	Location	DOE Comment	WSRC Response
1	DSA: Acronyms & Abbrev.	All	Correct the following inconsistencies in the Table: Add: Ci, DCF, HPFT, MACCS, MOI, NFPA, TNT, TPB, WSMS Delete: ASA, HA, LCO	Agree. Will correct Acronym and Abbreviation Table
2	DSA: E.4	2nd paragraph	In addition to the discussion of the Saltstone WAC AC, the Transfer Control AC to prevent inadvertent transfers from Tank 50H should be discussed.	Agree. Will add discussion on the Transfer Control AC
3	DSA: Chapter 2 TOC	Page 2-iv	Correct the page number for Figure 2.11-6 (should be page 2-21)	Agree. Will correct
4	DSA: 3.3	1st and 3rd paragraph	Spell out Consolidated Hazard Analysis (CHA) in the first paragraph rather than in the third paragraph	CHA is first spelled out on page ES-3. In the 3rd paragraph of section 3.3, CHA is spelled out because it is part of the title of a reference.
5	DSA: 3.4.5	All	Confirm that the BDBA dose to the public is correct (564 mrem) given a 6054 gallon release (by explosion) in Rev 1, versus a 16,181 gal release by fire in Rev 0 with a resulting <10 mrem dose.	The information is correct as written. The reason for the apparent discrepancy is two-fold. First, Rev 0 used the 2-hour DCF for a fire event (3.4E-05 rem/gal) while Rev 1 used the 3-minute DCF (8.65E-05 rem/gal) associated with a tank explosion event. Secondly, even though Rev 0 BDBA assumed that 60K gal of solution boiled, an ARF*RF of 2.0E-03 was applied (per DOE-HDBK-3010-94) that resulted in a ST of 120 gal. For the BDBA in Rev 1, we conservatively assumed that the ST was the entire contents of the SFT (6504 gal).

				For Rev 0: 120 gal x 3.4E-05 rem/gal = 4.08 mrem For Rev 1: 6504 gal x 8.65E-05 rem/gal = 563 mrem
6	DSA: Tables 3.6.4 & 3.6.5	All	Show the correct shading to correspond with the key.	Agree. Will correct
7	DSA: Table 3.6.6	All	Confirm that the bounding radionuclide concentrations (Ci/gal and rem/gal) used in Rev. 1 are the same as those used in Rev. 0.	The bounding radionuclide concentrations used in Rev. 0 and Rev. 1 are the same. The rem/gal information was not provided in Rev. 0, but the same bounding radionuclide concentrations were used for accident analysis in both revisions.
8	DSA: Table 3.6-9	Event 7.5.1	Radiation event where facility worker remains near vault for 8-hours. Does the consequence reflect the dose if worker were on top of the vault?	No, the initial consequence was developed for a worker near the vault. Event will be revised to include worker on the vault and the consequence (based on maximum expected dose on the vault of 1,720 mrem/hour) will be raised from "Neg." to "Low"
9	DSA: 6.4.5	2nd sentence	Correct font on "WSRC policy for"	Agree. Will correct.
10	DSA: 6.5.4.3	4th paragrap h	"Saltstone Facility WAC" should be replaced with "WSRC Procedure Manual 1S"	Agree. Will correct.
11	DSA: 6.7.4.4	2nd paragrap h	Procedure Manual "2Q2" should be replaced with "2Q"	The 2Q2 manual is specific to the Saltstone Facility and should be retained as a reference.
12	TSR: 5.6		Why were Safety Management Policies and Programs eliminated?	Safety Management Policies and Programs will be reinserted.