R10321136

ISSUED BY
RPP-WTP PDC
BM 8/25/04
INIT DATE

Document title: **Safety Requirements Document Volume II**

Contract number: DE-AC27-01RV14136

Department: Environmental and Nuclear Safety

Document number: 24590-WTP-SRD-ESH-01-001-02, Rev 3j

Date of issue: August 17, 2004

Issue status: Approved

Author(s): K Gibson

Principal author signature: K.D. Gibson

Checked by: J Duke

Checker signature: John Duke

Approved by: F Beranek

Approver's position: Environmental and Nuclear Safety Manager

Approver signature:

History Sheet

Rev	Date	Reason for revision	Revised by
0	21 Sep 2001	Supersedes BNFL-5193-SRD-01-02 Rev 5. Incorporates 24590-WTP-ABCN-ESH-01-017 Rev 0 changes (contractor-approved ABCN).	K Gibson
0a	4 Oct 2001	Incorporates ABCN-24590-01-00006 Rev 0 changes as partially approved by DOE Letter 01-OSR-0311 (CCN 023253) and contractor approved as revised ABCN-24590-01-00006, Rev 1.	K Gibson
0b	19 Feb 2002	Incorporates ABCN-24590-01-00004 Rev 2 changes as approved (with editorial changes specified) by DOE Letter 02-OSR-0034 (CCN 028170).	K Gibson
0c	20 Feb 2002	Incorporates 24590-WTP-ABCN-ESH-01-013 Rev 1 as approved by DOE Letter 02-OSR-0019 (CCN 028213).	K Gibson
0d	6 Mar 2002	This revision makes minor corrections to changes incorporated at Revisions 0b and 0c.	K Gibson
1	6 Jun 2002	Incorporates 24590-WTP-ABCN-ESH-01-015 Rev 0, 24590-WTP-ABCN-ESH-01-025 Rev 0, 24590-WTP-ABCN-ESH-01-027 Rev 0, 24590-WTP-ABCN-ESH-01-031 Rev 0, and 24590-WTP-ABCN-ESH-02-004 Rev 0 as approved by DOE Letter 02-OSR-0179 (CCN 033246); additionally, minor editorial changes were made throughout the document.	K Gibson
1a	24 Jun 2002	Incorporates ABCN-24590-01-00007 Rev 0, 24590-WTP-ABCN-ESH-01-022 Rev 0, 24590-WTP-ABCN-ESH-02-007 Rev 0, and 24590-WTP-ABDN-ESH-02-008 Rev 0 as approved by DOE Letter 02-OSR-0209 (CCN 034856). Revision bars from Revision 0d were incorrectly retained in Revision 1 to section 7.8. They have been removed at this revision to correct the error.	K Gibson
1b	25 Jun 2002	Incorporates 24590-WTP-ABCN-ESH-01-009 Rev 0 and 24590-WTP-ABCN-ESH-02-010 Rev 0 as approved by DOE Letter 02-OSR-0245 (CCN 035098).	K Gibson
1c	8 Jul 2002	Incorporates 24590-WTP-ABCN-ESH-01-003 Rev 0, 24590-WTP-ABCN-ESH-01-008 Rev 1, 24590-WTP-ABCN-ESH-02-011 Rev 0, and 24590-WTP-ABCN-ESH-02-013 Rev 0 as approved by DOE Letter 02-OSR-0232 (CCN 035603).	K Gibson
1d	26 Jul 2002	Incorporates 24590-WTP-ABCN-ESH-01-004 Rev 1 and 24590-WTP-ABCN-ESH-02-001 Rev 0 as approved by DOE Letter 02-OSR-0287 (CCN 036248).	K Gibson
1e	9 Aug 2002	Incorporates 24590-WTP-ABCN-ESH-01-028 Rev 1 as approved by DOE Letter 02-OSR-0304 (CCN 036553).	K Gibson

1f	6 Sep 2002	Incorporates 24590-WTP-ABCN-ESH-02-005 Rev 0, as approved by DOE Letter 02-OSR-0375 (CCN 039873); 24590-WTP-ABCN-ESH-01-007 Rev 2, as approved by DOE Letter 02-OSR-0385 (CCN 040108); 24590-WTP-ABCN-ESH-02-012 Rev 0, as approved by DOE Letter 02-OSR-0382 (CCN 040106); 24590-WTP-ABCN-ESH-01-005 Rev 1, as approved by DOE Letter 02-OSR-0374 (CCN 039810); 24590-WTP-ABCN-ESH-01-006 Rev 1, as approved by DOE Letter 02-OSR-0374 (CCN 039810), and made a minor correction to the pagination of Appendix C.14.	K Gibson
1g	17 Sep 2002	Incorporates 24590-WTP-ABCN-ESH-02-003 Rev 0, as approved by DOE Letter 02-OSR-0377 (CCN 048103), 24590-WTP-ABCN-ESH-02-015 Rev 0, as approved by DOE Letter 02-OSR-0372 (CCN 041100) and; 24590-WTP-ABCN-ESH-02-018 Rev 0, as approved by DOE Letter 02-OSR-0437 (CCN 041625).	K Gibson
1h	30 Sep 2002	Incorporates 24590-WTP-ABCN-ESH-01-020 Rev 1 as approved by DOE Letter 02-OSR-0427 (CCN 042023), 24590-WTP-ABCN-ESH-01-002 Rev 1 as approved by DOE Letter 02-OSR-0445 (CCN 043209), and 24590-WTP-ABCN-ESH-01-021 Rev 2 as approved by DOE Letter 02-OSR-0421 (CCN 043203).	K Gibson
1i	7 Oct 2002	Incorporates 24590-WTP-ABCN-ESH-01-029 Rev 1 as approved by DOE Letter 02-OSR-0449 (CCN 043897), 24590-WTP-ABCN-ESH-01-001 Rev 1 as approved by DOE Letter 02-OSR-0449 (CCN 043897), and 24590-WTP-ABCN-ESH-02-019 Rev 0 as partially approved by DOE Letter 02-OSR-0449 (CCN 043897).	K Gibson
1j	31 Oct 2002	Incorporates 24590-WTP-ABCN-ESH-02-026 Rev 0 as approved by DOE Letter 02-OSR-0490 (CCN 045060).	K Gibson
2	11 Dec 2002	Incorporates ABCN-24590-01-006, Rev 0 as approved by DOE Letter 02-OSR-0564 (CCN 046934), 24590-WTP-ABCN-ESH-01-020 Rev 1 as approved by DOE Letter 02-OSR-0427 (CCN 042023) for SRD Safety Criteria 7.2-7, 24590-WTP-ABCN-ESH-02-011 Rev 0 as approved by DOE Letter 02-OSR-0232 (CCN 035603) for SRD Safety Criterion 4.1-2, and 24590-WTP-ABCN-ESH-01-001 Rev 1 as approved by DOE Letter 02-OSR-0449 (CCN 043897)	K Gibson
2a	16 Jan 2003	Incorporates 24590-WTP-ABCN-ESH-02-028, Rev 0 as approved by DOE Letter 02-OSR-0608 (CCN 049603) additionally, a minor editorial change was made to Safety Criterion 4.4-8.	K Gibson
2b	7 Mar 2003	Incorporates 24590-WTP-ABCN-ESH-02-019, Rev 0 as approved by DOE Letter 03-OSR-0044 (CCN 052327)	K Gibson
2c	12 Mar 2003	Incorporates 24590-WTP-ABAR-ENS-03-008, Rev 0 as approved by DOE Letter 03-OSR-0072 (CCN 052423)	K Gibson
2d	13 Mar 2003	Incorporates 24590-WTP-ABAR-ENS-02-006, Rev 0 as approved by DOE Letter 03-OSR-0080 (CCN 052950)	K Gibson
2e	18 Mar 2003	Incorporates 24590-WTP-ABAR-ENS-02-007, Rev 0 as approved by DOE Letter 03-OSR-0096 (CCN 053531)	K Gibson

2f	26 Mar 2003	Incorporates 24590-WTP-ABAR-ENS-02-009, Rev 0 as approved by DOE Letter 03-OSR-0085 (CCN 054382) and corrects Table 4-1 and 4-2 titles per DOE Letter 03-OSR-0044 (CCN 052327)	K Gibson
2g	18 April 2003	Incorporates 24590-WTP-ABCN-ESH-02-023, Rev 0 and 24590-WTP-ABCN-ESH-02-033, Rev 0, as approved by DOE Letters 03-OSR-0123 (CCN 054985) and 03-OSR-0145 (CCN 054986)	K Gibson
2h	25 June 2003	Incorporates: (1) 24590-WTP-ABAR-ENS-02-002, Rev 0, as conditionally approved by DOE Letter 03-OSR-0211 (CCN 062341); (2) 24590-WTP-ABAR-ESH-02-005, Rev 1, as conditionally approved by DOE Letter 03-OSR-0223 (CCN 062343); (3) and 24590-WTP-ABAR-ENS-03-008, Rev 1, as conditionally approved by DOE Letter 03-OSR-0072 (CCN 052423).	K Gibson
2i	6 August 2003	Incorporates 24590-WTP-ABAR-ENS-02-014, Rev 1, as approved by DOE Letter 03-OSR-0264 (CCN 065098)	K Gibson
2j	15 August 2003	Incorporates 24590-WTP-SE-ENS-03-053, Rev 0 as approved by DOE Letter 03-OSR-0259 (CCN 065131).	K Gibson
2k	15 September 2003	Incorporates 24590-WTP-SE-ENS-03-551, Rev 0 as approved by DOE Letter 03-OSR-0321 (CCN 068946).	K Gibson
3	27 October 2003	Incorporates 24590-WTP-ABAR-ENS-02-002, Rev 0, as conditionally approved by DOE Letter 03-OSR-0326 (CCN 070209), 24590-WTP-SE-ENS-03-032, Rev 0 as conditionally approved by DOE letter 03-OSR-0359 (CCN 071752), 24590-WTP-ABAR-ESH-02-033, Rev 0 as conditionally approved by DOE Letter 03-OSR-0145 (CCN 054986); 24590-WTP-SE-ENS-03-480, Rev 0 as approved by DOE Letter 03-OSR-0363 (CCN 071865), 24590-WTP-SE-ENS-03-0484, Rev 0 as conditionally approved by DOE Letter 03-OSR-0362 (CCN 071864), 24590-WTP-SE-ENS-03-081, Rev 0 as approved by DOE Letter 03-OSR-0374 (CCN 072053), and 24590-WTP-SE-ENS-03-478, Rev 0 as approved by DOE Letter 03-OSR-0362 (CCN 072055); additionally minor editorial changes were made throughout the document.	K Gibson
3a	21 November 2003	Incorporates 24590-WTP-SE-ENS-03-771, Rev 0, as approved by DOE Letter 03-OSR-0389 (CCN 072805); additional editorial changes were made to safety criterion 1.0-6 and 4.1-2 and to the headers of section C.18 and C.19.	K Gibson
3b	18 December 2003	Incorporates 24590-WTP-SE-ENS-03-852, Rev 0, as approved by DOE Letter 03-OSR-0435 (CCN 076368); miscellaneous editorial changes.	K Gibson
3c	7 January 2004	Incorporates 24590-WTP-SE-ENS-03-771, Rev 0, as approved by DOE Letter 03-OSR-0445 (CCN 077938).	K Gibson
3d	5 February 2004	Incorporates 24590-WTP-SE-ENS-03-545, Rev 0, as approved by DOE Letter 03-OSR-0443 (CCN 079880).	K Gibson

3e	11 March 2004	Incorporates 24590-WTP-SE-ENS-03-962, Rev 0, as approved by DOE Letter 04-AMWTP-004 (CCN 080163) and modified by DOE Letter 04-WTP-033 (CCN 083770); 24590-WTP-SE-ENS-03-1290, Rev 1, as approved by DOE Letter 04-AMWTP-024 (CCN 083755); and 24590-WTP-SE-ENS-03-368, Rev 1, as approved by DOE Letter 04-WTP-027 (CCN 083769).	K Gibson
3f	17 March 2004	Incorporates 24590-WTP-SE-ENS-03-789, Rev 0, as approved by DOE Letter 04-AMWTP-018 (CCN 082523).	K Gibson
3g	13 April 2004	Incorporates 24590-WTP-SE-ENS-03-1209, Rev 0, as approved by DOE Letter 04-WTP-026 (CCN 085402), conditional approval of ABAR 24590-WTP-SE-ENS-03-368, Rev. 1, per DOE Letter 04-WTP-027 (CCN 083769), and editorial in Safety Criterion 4.2-2.	K Gibson
3h	12 May 2004	Incorporates 24590-WTP-SE-ENS-03-907, Rev 2, as approved by DOE Letter 04-ESQ-034 (CCN 087728).	K Gibson
3i	20 May 2004	Incorporates 24590-WTP-SE-ENS-04-032, Rev 0, as approved by DOE Letter 04-WTP-088 (CCN 089336)	K Gibson
3j	17 August 2004	Incorporates 24590-WTP-SE-ENS-04-011, Rev 0, as approved by DOE Letter 04-WTP-116 (CCN 092947); and 04-WTP-170 (CCN 096033); 24590-WTP-SE-ENS-03-1209, Rev 0, as modified by DOE Letter 04-WTP-128 (CCN 092470); and minor editorial changes to Safety Criterion 1.05 and Appendix C.31.	K Gibson

Revision Status

Document Part	Title	Revision	Pages w/Tracked Revisions
Front Matter	N/A	3j	xii
1.0	Radiological, Nuclear and Process Safety Objectives	3j	1-3
2.0	Radiological and Process Standards	3b	N/A
3.0	Nuclear and Process Safety	3b	N/A
4.0	Engineering and Design	3	N/A
4.1	General Design	3j	4.1-3 through 4.1-5
4.2	Confinement Design	3g	N/A
4.3	Engineered Safety Systems	3i	N/A
4.4	Electrical and Mechanical Systems	3i	N/A
4.5	Fire Protection	3	N/A
5.0	Radiation Protection	3	N/A
6.0	Startup	3	N/A
7.0	Management and Operations	3	N/A
7.1	Management and Organization/Staffing	3	N/A
7.2	Training and Procedures	3	N/A
7.3	Quality Assurance Program	3	N/A
7.4	Unreviewed Safety Questions	3	N/A
7.5	Conduct of Operations	3	N/A
7.6	Maintenance	3b	N/A
7.7	Reporting and Incident Investigation	3h	N/A
7.8	Emergency Preparedness	3	N/A
8.0	Deactivation and Decommissioning	3	N/A
9.0	Documentation and Submittals	3	N/A
Appendix A	Implementing Standard for Safety Standards and Requirements Identification	3j	A-1, A-15, A-20, and A-21
Appendix B	Implementing Standard for Defense in Depth	3b	N/A
Appendix C	Implementing Standards	3h	N/A
Front Matter	N/A	3h	N/A
C.1	ISO 10007, Quality Management - Guidelines for Configuration Management	3	N/A

Document Part	Title	Revision	Pages w/Tracked Revisions
C.2	DOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities	3j	C.2-1 through C.2-7
C.3	ANSI/AISC N690, Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities	3	N/A
C.4	This tailoring has been removed.	3	N/A
C.5	This tailoring has been removed.	3	N/A
C.6	NFPA 801, Standard for Facilities Handling Radioactive Materials	3	N/A
C.7	ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures	3e	N/A
C.8	ACI 318, Building Code Requirements for Structural Concrete and Commentary	3	N/A
C.9	AISC M016, Manual of Steel Construction, Allowable Stress Design (ASD)	3	N/A
C.10	UBC 97, Uniform Building Code	3	N/A
C.11	This tailoring has been removed.	3h	N/A
C.12	IEEE-387, Standard Criteria For Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations	3	N/A
C.13	IEEE-741, Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations	3	N/A
C.14	DOE/RL-94-02, Hanford Emergency Management Plan	3	N/A
C.15	DOE Order 5480.19, Conduct of Operations Requirements for DOE Facilities	3	N/A
C.16	DOE Order 433.1, Maintenance Management Program for DOE Nuclear Facilities; and DOE Guide 433.1-1 Nuclear Facility Maintenance Management Program Guide for Use with DOE O 433.1	3	N/A
C.17	Implementing of Class 1E, IEEE standards	3i	N/A
C.18	IEEE-308, Criteria for Class 1E Power Systems for Nuclear Power Generating Stations	3a	N/A
C.19	IEEE-384, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits	3i	N/A
C.20	IEEE-338, Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems	3	N/A

Document Part	Title	Revision	Pages w/Tracked Revisions
C.21	IEEE-628, IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations	3i	N/A
C.22	IEEE-344, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	3b	N/A
C.23	IEEE-323, Qualifying Class 1E Equipment for Nuclear Power Generating Stations	3	N/A
C.24	IEEE-379, Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems	3	N/A
C.25	NUREG-0800, Standard Review Plan, Section 6.4, "Control Room Habitability System", Section II	3	N/A
C.26	ASME B31.3-1996, Process Piping	3e	N/A
C.27	DOE Guide 421.1-2, Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830	3	N/A
C.28	DOE Order 5480.20A, Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities - Attachment 2, References and Definitions	3	N/A
C.29	DOE Order 420.1A, Facility Safety	3j	C.29-2
C.30	IEEE-382, IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants	3	N/A
C.31	IEEE-497, IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations	3j	C.31-5
C.32	ISO American Petroleum Institute Standards	3g	N/A
Appendix D	Radiological Exposure Standards for the RPP-WTP Project	3	N/A
Appendix E	Reliability, Availability, Maintainability, and Inspectability (RAMI)	3	N/A
Appendix F	Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning	3	N/A
Appendix G	Ad Hoc Implementing Standard for Safety Analysis Reports	3	N/A
Appendix H	Ad Hoc Implementing Standard for Erosion/Corrosion and Assessments	3	N/A
Appendix I	Ad Hoc Implementing Standard for Project Integrated Safety Management Approach	3	N/A
Appendix J	Ad Hoc Implementing Standard for Startup	3	N/A
Appendix K	Facility Areas Not Requiring Automatic Fire Suppression Systems Based on High Radiation and Low Combustible Loading	3f	N/A

Document Part	Title	Revision	Pages w/Tracked Revisions
Appendix L	Ad Hoc Implementing Standard for Seismic Design of Pressure Vessels	3	N/A

CONTENTS

APPENDICES

Introduction to the SRD

The Safety Requirements Document (SRD) provides formal documentation of the safety requirements and standards resulting from the Hanford Tank Waste Immobilization and Treatment Plant (WTP) Project safety standards and requirements identification process. Structures, systems, and components (SSCs) that serve to provide reasonable assurance that the WTP facility can be operated without undue risk are classified as Important to Safety and are defined in SRD Safety Criterion 1.0-6.

The process for establishing a set of radiological, nuclear, and process safety requirements and standards as described in DOE/RL-96-0004 and RL/REG-98-17 is referred to as the Integrated Safety Management (ISM) process. For specific SRD safety criteria implementing codes and standards are specified for the safety design class, safety design significant, safety class, and safety significant SSCs. For specific SRD safety criteria implementing codes and standards for risk reduction class (RRC) and additional protection class (APC) SSCs shall be specified using the process set forth in this SRD Appendix A ISM process (i.e., the implementing standard for safety standards and requirements identification to meet DOE/RL-96-0004) and need not otherwise be specified in the SRD with one exception: For appendices in the SRD designated as "implementing standards" provisions of these appendices specified for RRC and APC SSCs remain in effect. This paragraph is only applicable to the following Safety Criteria: 4.1-1, 4.1-2, 4.1-3, 4.1-4, 4.2-1, 4.2-2, 4.2-3, 4.3-1, 4.3-2, 4.3-3, 4.3-4, 4.3-5, 4.3-6, 4.3-7, 4.4-1, 4.4-2, 4.4-3, 4.4-4, 4.4-5, and 4.4-6. However, for Safety Criteria 4.1-2 and 4.1-3, the implementing codes and standards contained in these safety criteria shall be applicable to APC SSCs designated SC-II or SC-III as they apply to seismic performance.

1.0 Radiological, Nuclear and Process Safety Objectives

Safety Criterion: 1.0 - 1

Applicable Project Phases - All

A comprehensive process safety management program shall be used to eliminate or reduce the incidence, or mitigate the consequences of, accidental hazardous chemical releases, process fires, and process explosions. This program shall address management practices, technologies, and procedures. Process safety management shall confirm that the facility is properly designed, the integrity of the design is maintained, and the facility is operated according to the safe manner intended.

Implementing Codes and Standards

Regulatory Basis

DOE/RL-96-0006	*5.1.1*	*Process Safety Management*
DOE/RL-96-0006	*5.1.2*	*Process Safety Objective*

Safety Criterion: 1.0 - 2

Applicable Project Phases - All

The risk, to an average individual in the vicinity of the Contractor's facility, of prompt fatalities that might result from an accident shall not exceed one-tenth of one percent (0.1 %) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population generally are exposed. (For evaluation purposes, individuals are assumed to be located within 1 mile of the controlled area.)

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix D, "Radiological Exposure Standards for the WTP Project"

Regulatory Basis

DOE/RL-96-0006	*3.1.2*	*Accident Risk Goal*

Safety Criterion: 1.0 - 3

Applicable Project Phases - All

The risk, to the population (public and workers) in the area of the Contractor's facility, of cancer fatalities that might result from facility operation shall not exceed one-tenth of one percent (0.1 %) of the sum of cancer fatality risks to which members of the U.S. population generally are exposed. (For evaluation purposes, individuals are assumed to be located within 10 miles of the controlled area.)

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix D, "Radiological Exposure Standards for the WTP Project"

Regulatory Basis

DOE/RL-96-0006 *3.1.1* *Operations Risk Goal*

Safety Criterion: 1.0 - 4

Applicable Project Phases - All

This safety criterion has been deleted.

Safety Criterion: 1.0 - 5

Applicable Project Phases - All

To compensate for potential human and equipment failures, a defense-in-depth strategy shall be applied to the facility commensurate with the hazards; such that, as appropriate to control the risk, safety is vested in multiple, independent safety provisions, no one of which is to be relied upon excessively to protect the public, the workers, or the environment. This strategy shall be applied to the design and operation of the facility. Consistent with the defense-in-depth principle, the WTP will be designed with the objective of providing multiple layers of protection to prevent or mitigate the unintended release of radioactive materials to the environment. These multiple layers of protection shall include the following:

- Principle emphasis shall be placed on the prevention of accidents, particularly any that could cause an unacceptable release, as the primary means of achieving safety. Prevention of accidents shall be provided through measures such as siting to alleviate the need to provide design measures; minimizing and controlling the material at risk; and providing a conservative design such that a significant margin exists between the design limit and the ultimate failure point of safety structures, systems, and components. The single failure criterion shall be applied in a manner proportionate to the magnitude and nature of the hazard.
- Controls on normal operations, including anticipated operational occurrences, maintenance, and testing, so that facility and system variables remain within their operating ranges and the frequency of demands placed on structures, systems, and components important to safety is small.
- Conservatively designed confinement systems retain and mitigate the radioactive materials associated with the entire range of events considered in the design basis. The confinement systems should protect the workplace and the environment. The confinement systems shall be capable of satisfying the standards in SRD section 2.0 with margin for all operational occurrences and all events considered in the design basis events.
- Automatic systems to restrict deviations from normal operations, to place and maintain the facility in a safe state, and to limit the potential spread of radioactive materials when operating limits exceed predetermined setpoints. Operator actions may also perform these functions. Operator actions may be credited only if analysis demonstrates that the total time interval required to perform the operator action is shorter than the time at which the limiting design requirement would be reached without operator action.
- The human aspects of defense in depth including a design for human factors, a quality assurance program, administrative controls, internal safety reviews, operating limits (Technical Safety Requirements), worker qualification and training, and establishment of a safety/quality program.

- Preparation for emergencies by providing emergency preparedness plans and by making provision to monitor accident releases as necessary to support emergency responses.

The number of confinement barriers and other controls provided against a particular hazard shall be a function of the potential consequences from the hazard. This will result in provision of a level of control tailored to the significance of the hazard. Adequate defense in depth shall be confirmed by accident analyses that show that the exposure standards in SRD Section 2.0 are met with margin and by risk analyses that show that the risk goals in SRD Section 3.0 are satisfied.

Implementing Codes and Standards

ANSI/ANS 58.8-1994, *Time Response Design Criteria for Safety-Related Operator Actions*
ANSI/ANS 58.9-1981, *Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems*
ASME B31.3-96, *Process Piping*
ASME SEC VIII, *Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels*
DOE G 420.1-1, Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria Guide for use with DOE O 420.1, Facility Safety, Section 2.3
DOE Order 420.1A, *Facility Safety,* Section 4.1.1.2
IEEE 379-1994, *Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems,* as tailored in Appendix C
IEEE 1023-1988, *IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations*
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II* Appendix B, "Implementing Standard for Defense in Depth"

Regulatory Basis

DOE/RL-96-0006	*4.1.1.1*	*Defense in Depth-Defense in Depth*
DOE/RL-96-0006	*4.1.1.2*	*Defense in Depth-Prevention*
DOE/RL-96-0006	*4.1.1.3*	*Defense in Depth-Control*
DOE/RL-96-0006	*4.1.1.4*	*Defense in Depth-Mitigation*
DOE/RL-96-0006	*4.1.1.5*	*Defense in Depth-Automatic Systems*
DOE/RL-96-0006	*4.1.1.6*	*Defense in Depth-Human Aspects*
DOE/RL-96-0006	*4.2.1.1*	*Design-Safety Design*

Safety Criterion: 1.0 - 6

Applicable Project Phases - All

Important to safety structures, systems, and components shall be identified and sub-classified as safety-class, safety-significant, and additional-protection class. SSCs currently classified as safety design class, safety design significant, and RRC shall remain under the SDC/SDS/RRC classification method until reclassified using the contract-approved change process.

Important to safety: Structures, systems, and components (SSCs) that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public are classified as Important to Safety. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological, nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation).

This definition includes not only those structures, systems, and components that perform safety functions and traditionally have been classified as safety class, safety-related or safety-grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems, or components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear, and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of Important to Safety, i.e., safety-related may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems, and components should be considered for inclusion within this definition. [DOE/RL-96-0006]

Safety structures, systems, and components means both safety-class structures, systems, and components and safety-significant structures, systems, and components. [10 CFR 830]

Safety-class structures, systems, and components (SC SSC) means the structures, systems, or components, including portions of process systems, whose preventive or mitigative function is necessary to limit radioactive hazardous material exposure to the public, as determined from safety analyses. [10 CFR 830]

Safety-significant structures, systems, and components (SS SSC) means the structures, systems, and components which are not designated as safety-class structures, systems, and components, but whose preventative or mitigative function is a major contributor to defense in depth and/or worker safety as determined from safety analyses. [10 CFR 830]

Additional-protection class structures, systems, and components (APC SSC) means the structures, systems, and components important to safety that are neither safety-class nor safety-significant.

Safety Design Class (SDC). Safety Design Class SSCs are the following:

(1) SSCs whose safety function is to prevent a worker or the maximally exposed member of the public from receiving a radiological exposure that exceeds the exposure standards defined in the SRD;

(2) SSCs whose safety function is to prevent a worker or the maximally exposed member of the public from receiving a chemical exposure that exceeds the exposure standards defined in the SRD; or

(3) SSCs credited for the prevention of a criticality event.

Safety Design Significant (SDS). Safety Design Significant SSCs are the following:

(1) SSCs that are required to ensure that exposure standards for normal operation are not exceeded;

(2) SSCs whose failure would directly prevent Safety Design Class SSCs from performing their safety function; or

(3) SSCs that are required to meet SRD Appendix B, section 3.0, Table 1, "Implementation of Defense in Depth by SSCs."

Risk Reduction Class (RRC). RRC SSCs are Important to Safety SSCs that are neither SDC nor SDS. For example, an SSC that is neither SDC nor SDS and whose function is necessary to ensure the integrity of boundaries retaining radioactive materials, is classified as RRC only when the SSC contains a significant amount of radioactivity.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix D, "Radiological Exposure Standards for the WTP Project"

Regulatory Basis

DOE/RL-96-0006	*3.3.1*	*Public Protection*
DOE/RL-96-0006	*3.3.2*	*Worker Protection*
10 CFR 830		

Safety Criterion: 1.0 - 7

Applicable Project Phases - All

The WTP Contractor shall accept responsibility for the safety of the WTP. In no way shall this responsibility be diluted by the separate activities and responsibilities of designers, suppliers, constructors, the Safety Regulation Division (OSR), or independent oversight bodies. This responsibility shall be exerted through a strong, unambiguous organizational structure. The assignment and subdivision of responsibility for safety shall be kept well defined throughout the life of the facility.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02. *Safety Requirements Document Volume II,* Appendix I - "Ad Hoc Implementing Standard for Project Integrated Safety Management Approach

Regulatory Basis

DOE/RL-96-0006	*4.1.2.1*	*Safety Responsibility-Safety Responsibility*
DOE/RL-96-0006	*4.1.2.2*	*Safety Responsibility - Safety Assignments*
DOE/RL-96-0006	*4.3.1.1*	*Conduct of Operations-Organizational Structure*
DOE/RL-96-0006	*5.1.3*	*Process Safety Responsibility*

2.0 Radiological and Process Standards

Safety Criterion: 2.0 - 1

Applicable Project Phases - All

The following radiological exposure standards shall be applied to protect the public and workers from WTP radiological hazards. See Figure 1 for Location of Facility and Co-located Workers and Figure 2 for the Boundary to Location for Offsite Receptor.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix D, "Radiological Exposure Standards for the WTP Project"

Regulatory Basis

DOE/RL-96-0006 *2.1* *Individual (Dose Standards Above Normal Background)*

Table 2-1 Radiological Exposure Standards Above Normal Background[4]

Description	Estimated Frequency of Occurrence f (yr^{-1})	General Guidelines	Facility Worker	Co-located worker	Public
<u>Normal Events</u> Events that occur regularly in the course of facility operation (e.g., normal facility operations); including routine and preventive maintenance activities.	>0.1	-----	≤5 rem/yr ≤50 rem/yr any organ, skin, or extremity ≤15 rem/yr lens of eye ≤1.0 rem/yr ALARA design objective per 10CFR835.1002(b)[(1)]	≤5 rem/yr ≤1.0 rem/yr ALARA design objective per 10 CFR 835.1002 (b)[(1)]	≤1.5 mrem/yr (airborne pathway[5]) ≤100 mrem/yr (all sources) ≤100 mrem/yr (public in the controlled area) ≤25 mrem/yr (radioactive waste)
<u>Anticipated Events</u> Events of moderate frequency that may occur once or more during the life of a facility (e.g., minor incidents and upsets).	$10^{-2}<f\leq10^{-1}$	-----	≤5 rem/event [(2)] 1.0 rem/event design action threshold [(3)]	≤5 rem/event[(2)] 1.0 rem/event design action threshold[(3)]	≤100 mrem/event[(2)]

Table 2-1 Radiological Exposure Standards Above Normal Background[4]

Description	Estimated Frequency of Occurrence f (yr^{-1})	General Guidelines	Facility Worker	Co-located worker	Public
Unlikely Events Events that are not expected, but may occur during the lifetime of a facility (e.g., more severe incidents).	$10^{-4}<f\leq10^{-2}$	-----	≤25 rem/event(2)	≤25 rem/event(2)	≤5 rem/event(2)
Extremely Unlikely Events Events that are not expected to occur during the life of the facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.	$10^{-6}<f\leq10^{-4}$	-----	≤100 rem/event(2)	≤25 rem/event(2)	≤25 rem/event ≤5 rem/event target(2) ≤300 rem/event to thyroid
Location of Receptor			Within the WTP Controlled Area Boundary	The most limiting location at or beyond the WTP Controlled Area Boundary	The most limiting location along the near river bank/ Hwy240/ southern boundary

Notes

1 In addition to meeting the listed design objective of 10 CFR 835.1002(b), the inhalation of radioactive material by workers and co-located workers under normal conditions is kept ALARA through the control of airborne radioactivity as described in 10 CFR 835.1002(c).

2 In addition to meeting the listed exposure standards for accidents, the approach to accident mitigation is to evaluate accident consequences to ensure that the exposures are far enough below standards to account for uncertainties in the analysis, and to provide for sufficient design margin and operational flexibility.

3 When a calculated accident exposure exceeds this threshold, then appropriate actions are taken. These include carrying out a less bounding (i.e., more realistic) evaluation to show that the accident consequences will be below the threshold or evaluating additional safeguards for cost-effectiveness and/or feasibility. This threshold is not a limit; it does not require the implementation of additional preventative or mitigative features if they are not both cost-effective and feasible.

4 The dose terms presented in this Table are defined in 10 CFR 835.

5 The dose value for the "public" airborne pathway is calculated in accordance with Safety Criterion 5.1-2.

Figure 2-1 Location of Facility and Co-located Workers

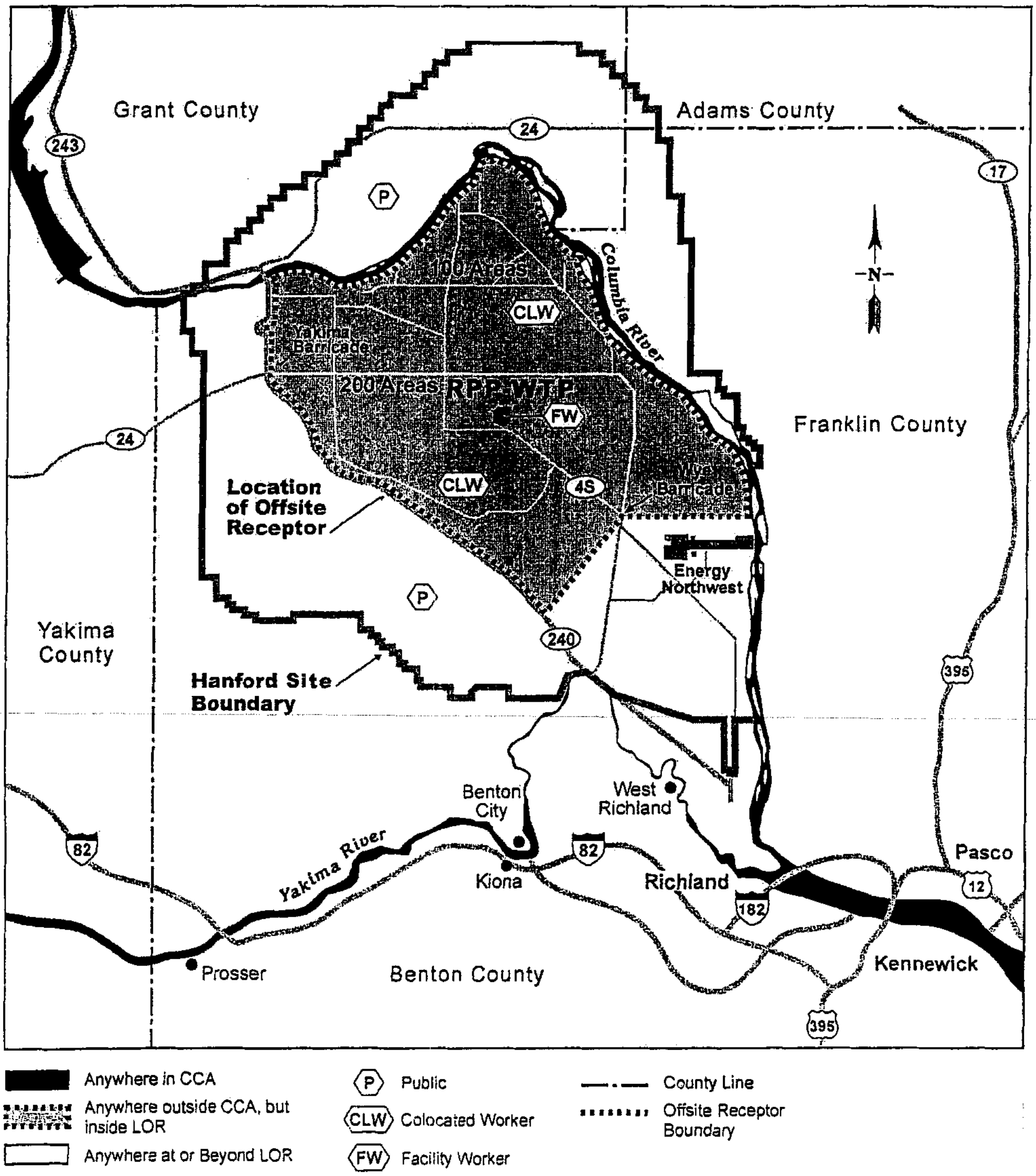
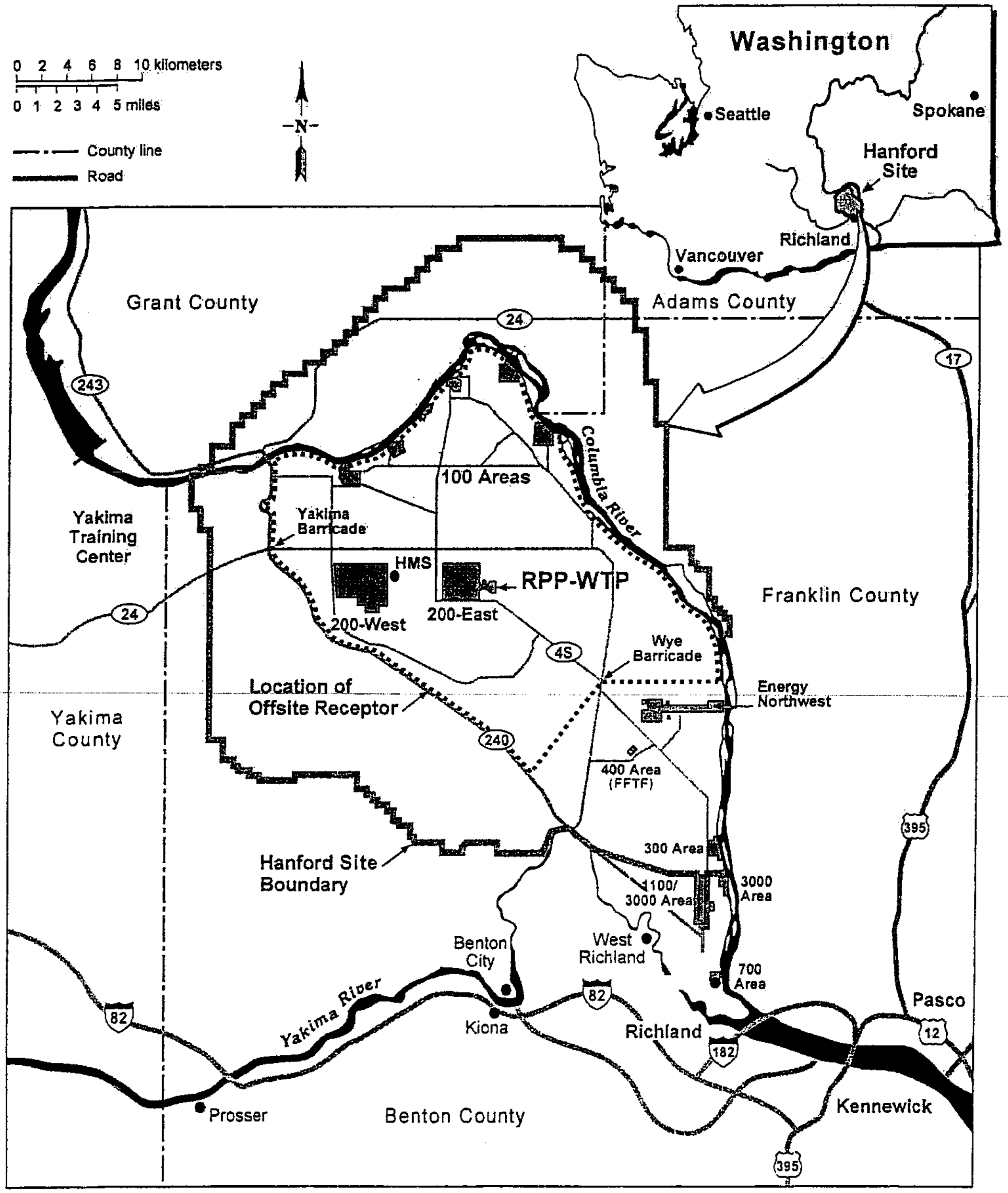


Figure 2-2 Boundary for Location of Offsite Receptor for the Purpose of Implementing DOE/RL-96-0006, Rev. 0, Table 1, Public Exposure Standard



Safety Criterion: 2.0 – 2

Applicable Project Phases - All

The following dose standards shall be applied to protect the public and workers from WTP chemical hazards.

- Releases exposing the offsite public to 2001 American Industrial Hygiene Association (AIHA) Emergency Response Planning Guideline—2 (ERPG-2) concentrations.
- Releases exposing the co-located worker to 2001 AIHA ERPG-3 concentrations.
- Accidents affecting the facility worker that could cause in-patient hospitalization of at least 3 facility workers, or at least a single fatality.
- Where ERPG values have not been published, the 2001 DOE Temporary Emergency Exposure Limits (TEELs) Revision 17m shall be used as substitute ERPGs.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006 *5.1.1* *Process Safety Management*

Safety Criterion: 2.0 – 3

Applicable Project Phases - All

In addition to the dose limits specified for the public in Safety Criterion 2.0-1 Table 2-1, the dose in any unrestricted area from external sources shall not exceed 0.002 rem in any one hour.

Implementing Codes and Standards

DOE G 441.1-2, *Occupational ALARA Program Guide*

Regulatory Basis

WAC 246-221 *Radiation Protection Standards* *Location: 060 (1)*

3.0 Nuclear and Process Safety

3.1 Hazards Analysis

Safety Criterion: 3.1 - 1

Applicable Project Phases - All

A process hazard analysis (PHA) shall be performed using acceptable industry practices. The PHA method used shall be appropriate to the complexity of the processes and the associated chemical and radiological hazards. The PHA, with the standards selection and accident analyses processes, confirms the adequacy of the design and operation controls provided to protect the facility workers, co-located workers, the public, and the environment. The PHA shall be performed at the earliest practical point in conceptual or preliminary design, so that required functional attributes of ITS SSCs can be specified in the detailed design. The PHA shall be based upon the identification of work, which includes definition of the project mission and identification of the processes that must be performed to accomplish the mission so that the hazards inherent in the work can be identified. Process safety information shall be compiled pertaining to the hazards of the materials used or produced by the process, the technology of the process, and the equipment in the process. Identification of work for the purpose of design development involves definition of various plant systems, structures, and components. The PHA shall consider the effects of engineered and administrative controls and the consequences of their failure, human factors, facility siting, common-mode and common-cause failure events, and previous incidents. The analysis shall evaluate the adequacy of the design and operating procedures. The analyses shall initially consider the hazardous situation as being unmitigated (credit may be taken for passive features not challenged by the situation) and then evaluate the adequacy of the design and operating procedures to prevent or mitigate the event.

The PHA shall be performed by teams that include expertise in engineering, process operations, the process being evaluated, and the specific process hazard analysis methodology being used.

The results of the PHA shall be documented including process hazards and possible safety, health, and environmental effects. A system shall be established to address and document the PHA findings in order to assure that the findings are resolved and that the equipment and procedures provide an adequate degree of protection against accidents. The contractor shall document what actions are to be taken; complete actions; develop a written schedule of when these actions are to be completed; communicate the actions to operating maintenance and other employees whose work assignments are in the process and who may be affected by the recommendations or actions. The PHA shall be updated concurrently with the annual update of the Safety Analysis Report to ensure that the process hazard analysis is consistent with the current process.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*

Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

AICHE, 1999, *Guidelines for Hazard Evaluation Procedures*, American Institute of Chemical Engineers, New York, NY

Regulatory Basis

DOE/RL-96-0006	*3.3.3*	*Accident Vulnerability Mitigation*
DOE/RL-96-0006	*5.1.1*	*Process Safety Management*
DOE/RL-96-0006	*5.1.2*	*Process Safety Objective*
DOE/RL-96-0006	*5.2.1*	*Process Safety Information*
DOE/RL-96-0006	*5.2.2*	*Process Hazard Analysis*

Safety Criterion: 3.1 - 2

Applicable Project Phases - All

A compilation of written process safety information appropriate to the stage of design being considered shall be completed to support the process hazard analysis. The compilation of written process safety information enables the employer and the employees involved in operating the process to identify and understand the hazards posed by those processes involving radioactive materials and process chemicals considered to pose a hazard. This process safety information shall include information pertaining to the hazards of the materials used or produced by the process, information pertaining to the technology of the process, and information pertaining to the equipment in the process.

(1) Information pertaining to the hazards of the materials in the process including:
- (a) Toxicity information
- (b) Permissible exposure limits
- (c) Physical data
- (d) Reactivity data
- (e) Corrosivity data
- (f) Thermal and chemical stability data
- (g) Hazardous effects of inadvertent mixing of different materials that could foreseeably occur

(2) Information pertaining to the technology of the process including at least the following:
- (a) A block flow diagram or simplified process flow diagram
- (b) Process chemistry
- (c) Maximum intended inventory
- (d) Safe upper and lower limits for such items as temperatures, pressures, flows or compositions
- (e) An evaluation of the consequences of deviations, including those affecting the safety and health of employees

(3) Information pertaining to the equipment in the process including:
- (a) Materials of construction
- (b) Process drawings or piping and instrument diagrams (P&IDs)
- (c) Electrical classification
- (d) Relief system design and design basis

(e) Ventilation system design

(f) Design codes and standards employed

(g) Material and energy balances

(h) Safety systems (e.g., interlocks, detection or suppression systems)

The records shall be maintained documenting that equipment complies with recognized and generally accepted good engineering practices. The safety information shall be kept up-to-date.

Implementing Codes and Standards

Regulatory Basis

DOE/RL-96-0006	*5.2.1*	*Process Safety Information*
DOE/RL-96-0006	*5.2.2*	*Process Hazard Analysis*

3.2 Accident Analysis

Safety Criterion: 3.2 - 1

Applicable Project Phases - All

Accident analyses shall be performed that assist in the identification of accident prevention and mitigation SSCs, the establishment of the safety functions and performance requirements of the identified SSCs, the selection of standards necessary to ensure the safety and performance requirements of the SSCs are achieved, and the development of the emergency preparedness program. Particular care will be taken to identify, evaluate, and prevent and/or mitigate any vulnerabilities to accidents that might by themselves, result in a release of radioactive material that exceed acceptable levels. Measures in the design and operation of the facility to protect the facility and co-located workers and the public against accident conditions should be evaluated using an acceptable approach to demonstrate that they perform their intended purpose with high confidence.

The accident analyses, with the process hazard analysis and standards selection process, confirm the adequacy of the controls provided to protect the facility and co-located workers, the public, and the environment. The accident analyses shall also demonstrate the adequacy of confinement barriers to effectively perform their required functions. Accident analyses shall consider facility hazards; hazardous situations (accidents) resulting from normal operation, anticipated occurrences, maintenance, and testing; natural phenomena hazards; and external man-induced hazards. An accident analysis shall be performed at the earliest practical point in conceptual or preliminary design, so that required functional attributes of safety SSCs can be specified in the detailed design.

Compliance with radiological exposure standards for facility workers may use qualitative methods, supported by numerical analysis, where necessary.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006	*3.3.1*	*Public Protection*
DOE/RL-96-0006	*3.3.2*	*Worker Protection*
DOE/RL-96-0006	*3.3.3*	*Accident Vulnerability Mitigation*
DOE/RL-96-0006	*4.2.1.1*	*Safety Design*
DOE/RL-96-0006	*4.2.1.2*	*Risk Asssessment*
DOE/RL-96-0006	*4.2.1.3*	*Safety Analysis*
DOE/RL-96-0006	*5.1.1*	*Process Safety Management*
DOE/RL-96-0006	*5.1.2*	*Process Safety Objective*
DOE/RL-96-0006	*5.2.1*	*Process Safety Information*
DOE/RL-96-0006	*5.2.2*	*Process Hazard Analysis*

Safety Criterion: 3.2 - 2

Applicable Project Phases - All

Hazard control strategies in terms of design and administrative controls shall be identified to manage by prevention or mitigation potential accidents such that compliance to the radiological and chemical exposure standards of SC 2.0-1 and 2.0-2 and protection of the environment are provided. Selection of the hazard control strategies may require iteration with the hazard analysis (SC 3.1-1) and the standards selection process (SC 3.2-3), and will result in the facility being designed for a set of events such as: normal operation, including anticipated operational occurrences, maintenance, and testing; external events; and postulated accidents.

Consistent with the defense in depth principle, the control strategy development should emphasize accident preventive measures over mitigative measures. It should also emphasize passive structures, systems, and components (SSCs) over active SSCs and retention of released material over dispersion.

Significant new design features should be introduced only after thorough research and model or prototype testing at the component, system, or facility level, as appropriate.

Hazard control strategies shall be evaluated for the most bounding conditions (i.e., the most demanding requirements imposed by the set of hazardous situations that credit the function of the hazard control strategy). In addition, the evaluation of the hazard control strategy shall identify the performance requirements (including environmental conditions) necessary to assure that it performs its functions reliably. Such measures include maintenance requirements, testing intervals and calibration frequency.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standards for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006	*4.2.1.1 Safety Design*
DOE/RL-96-0006	*4.2.2.1 Proven Engineering Practices*
DOE/RL-96-0006	*4.2.2.3 Safety System Design and Qualification*
DOE/RL-96-0006	*4.2.5 Inherent/Passive Safety Characteristics*

Safety Criterion: 3.2 - 3

Applicable Project Phases - All

Design codes and standards are identified or developed, evaluated, and tailored to provide assurance that the hazard control strategies identified by SC 3.2-2 will perform their specified accident prevention or mitigation function when called upon. Standards are also developed to provide for compliance with applicable laws and regulations and conformance with the DOE-stipulated top-level standards. Accident prevention and mitigation safety technologies incorporated into the facility design shall have been proven by experience or testing and shall be reflected in approved design codes and standards.

Documentation of the standards development process provides justification of the set of selected standards developed and links hazard control strategies to their associated set of design codes.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006 *4.2.2.1 Proven Engineering Practices*

3.3 Criticality

Safety Criterion: 3.3 - 1

Applicable Project Phases - All

The facility shall be designed and operated in a manner that prevents nuclear criticality and that complies with the requirements of DOE Order 420.1A (DOE O 420.1A), section 4.3, "Nuclear Criticality Safety".

Implementing Codes and Standards

DOE O 420.1A, *Facility Safety*

Safety Criterion: 3.3 - 2

Applicable Project Phases - All

The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and in the nature of the immediate environment under accident conditions.

The multiplication factor, k-eff, as calculated by a method to demonstrate subcriticality (e.g., MCNP calculation) shall be less than 1.0 by an amount that includes a 5 % Minimum Subcritical Margin (MSM). In formula form, this criterion is expressed as follows:

Keff < 1.0 - (MSM + 2(sigma) + AoA) + (code bias - code bias uncertainty)

Here the MSM is defined to be a conservative factor on top of all the other margins listed in the equation. The code bias is quantified in the code validation that statistically compares results from computations to critical experiments. In quantifying the calculational bias (code bias - code bias uncertainty), the associated bias uncertainty is also included. The sum of these two values can be either positive or negative. If positive, they are to be set to zero. Thus only negative values of the sum of code bias and code bias uncertainty are included in the above equation.

The factor 2(sigma) is the statistical uncertainty of the calculational method at 95 % confidence interval. The AoA Margin is also determined during the code validation process. The comparisons for the code validation attempt to select critical experiments that have characteristics similar to those modeled in the CSER computations. The selection of a group of experiments is justified by an AoA comparison in which the important neutronic parameters are demonstrated within the same ranges or AoA for both the critical experiments and the CSER computations.

4.0 Engineering and Design

Safety Criterion: 4.0 – 1

Applicable Project Phases - All

Formal configuration management shall be applied to all facility activities through deactivation of the WTP to ensure that programmatic objectives, including safety, are fully achieved. Work shall be performed and controlled according to pre-approved plans and procedures that clearly delineate responsibility. Documented records shall be retained.

Implementing Codes and Standards

ISO 10007:1995(E), *Quality Management - Guidelines for Configuration Management,* as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006 *4.1.5.1 Configuration Management-Formal Configuration Management*

Safety Criterion: 4.0 – 2

Applicable Project Phases - All

Written procedures shall be established and implemented to manage changes (except for "replacements in kind") to process chemicals, technology, equipment, and procedures; and, changes to facilities that affect a covered process. The procedures shall assure that the following considerations are addressed prior to any change:

(1) The technical basis for the proposed change

(2) Impact of change on safety and health

(3) Modifications to operating procedures

(4) Necessary time period for the change

(5) Authorization requirements for the proposed change

Employees involved in operating a process and maintenance and subcontract employees whose job tasks will be affected by a change in the process shall be informed of, and trained in, the change prior to start-up of the process or affected part of the process. If a change covered by this paragraph results in a change in the process safety information, such information shall be updated accordingly. If a change covered by this paragraph results in a change in operating procedures or practices, such procedures or practices shall be updated accordingly.

Implementing Codes and Standards

ISO 10007:1995(E), *Quality Management - Guidelines for Configuration Management,* as tailored in Appendix C.

Regulatory Basis

DOE/RL-96-0006 *5.2.9 Management of Change*

Safety Criterion: 4.0 – 3

Applicable Project Phases - All

A system shall be used to control and maintain accurate as-built records for Important to Safety SSCs through deactivation of the facility.

Implementing Codes and Standards

ISO 10007:1995(E), *Quality Management - Guidelines for Configuration Management*, as tailored in Appendix C.

Regulatory Basis

DOE/RL-96-0006 *4.1.5.3 Configuration Management-Design Documentation*

4.1 General Design

Safety Criterion: 4.1 - 1

Applicable Project Phases - All

The facility design shall provide for the prevention and mitigation of the risks associated with radiological and chemical material inventories and energy sources. The facility design shall include consideration of normal operation (including startup, testing and maintenance), anticipated operational occurrences, external events, and accident conditions.

Prevention shall be the preferred means of achieving safety.

Defense-in-depth shall be applied commensurate with the hazard to provide multiple physical and administrative barriers against undue radiation and chemical exposure to the public and workers.

Implementing Codes and Standards

ANSI/ANS 58.9-1981, *Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems*
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix B, "Implementing Standard for Defense in Depth"
DOE G 420.1-1, *Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria Guide for Use with DOE O 420.1, Facility Safety*
Section 2.3
DOE Order 420.1A, *Facility Safety*
Section 4.1.1.2
IEEE 379-1994, *Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems,* as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006	*4.1.1.1*	*Defense in Depth-Defense in Depth*
DOE/RL-96-0006	*4.1.1.2*	*Defense in Depth-Prevention*
DOE/RL-96-0006	*4.2.1.1*	*Design-Safety Design*

Safety Criterion: 4.1 - 2

Applicable Project Phases - All

Structures, systems, and components designated as Important to Safety shall be designed, fabricated, erected, constructed, tested, inspected, and maintained to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components designated as Important to Safety shall be maintained through deactivation of the facility.

Items and processes shall be designed using sound engineering/scientific principles and appropriate standards.

Design features that enhance the margin of safety through simplified, inherently safe, passive, or other highly reliable means to accomplish the specified safety function should be employed to the maximum extent practical.

Design work, including changes, shall incorporate applicable requirements and design bases. Design interfaces shall be identified and controlled. The adequacy of design products shall be verified or validated by individuals or groups other than those who performed the work. Verification and validation work shall be completed before approval and implementation of the design.

Safety technologies incorporated into the facility design should have been proven by experience or testing and should be reflected in approved codes and standards. Significant new design features should be introduced only after thorough research and model or prototype testing at the component, system, or facility level, as appropriate, to achieve the necessary level of confidence that the design feature will perform as expected.

Implementing Codes and Standards

ACI 318-99, *Building Code Requirements for Structural Concrete*, as tailored in Appendix C.
ACI 318R-99, *Commentary on Building Code Requirements for Structural Concrete*
ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*, as tailored in Appendix C.
ACI 349R-01, *Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures*
AISC M016-89, *Manual for Steel Construction - Allowable Stress Design, Ninth Edition*, as tailored in Appendix C.
ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*, as tailored in Appendix C.
ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary*
ASCE 7-98, *Minimum Design Loads for Buildings and Other Structures*
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
1997, *UBC Uniform Building Code*, as tailored in Appendix C.
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98
ACI 530-99, *Building Code Requirements for Masonry Structures and Commentary*
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
ISO 10007:1995(E), *Quality Management - Guidelines for Configuration Management*, as tailored in Appendix C.
ASTM D3740, *Standard Practice for Minimum Requirements for Agencies Engaged in the Testing and/or Inspection of Soil and Rock as Used in Engineering Design and Construction*
ASTM D2922, *Standard Test Methods Density of Soil and Soil-Aggregate in Place by Nuclear Methods (Shallow Depth)*
ASTM D3017, *Standard Test Method for Water Content of Soil and Rock in Place by Nuclear Methods*
DOE-RL-92-36, Hanford Site Hoisting and Rigging Manual
CMAA 70-2000, Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes (supplemented with ASME NOG 1-2002, Sections NOG-1140, NOG-4150, NOG-5482, NOG-6120b and NOG-6150 for SDS cranes). Note: Seismic acceleration loads shall be included in the extraordinary loadings identified in CMAA 70-2000.
CMAA 74-2000, Specifications for Top Running and Under Running Single Girder Electric Overhead Traveling Cranes Utilizing Under Running Trolley Hoist (supplemented with ASME NUM 1-2000 [with NUM 1a-2002], Sections NUM-G2000, NUM-II-7000, NUM-II-8200, NUM-II-8300, and NUM-II-8400 for SDS cranes). Note: Seismic acceleration loads shall be included in the extraordinary loadings identified in CMAA 74-2000.
ASME NUM 1-2000 [with NUM 1a-2002 Addenda], Rules for Construction of Cranes: Monorails and Hoists (With Bridge or Trolley or Hoist of the Underhung Type)(For SDC cranes only)
ASME NOG 1-2002, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)(For SDC cranes only)

Regulatory Basis

DOE/RL-96-0006	*4.1.2.4*	*Safety Responsibility-Operating Experience and Safety Research*
DOE/RL-96-0006	*4.1.5.1*	*Configuration Management-Formal Configuration Management*

DOE/RL-96-0006	*4.1.6.2*	*Quality Assurance-Established Techniques and Procedures*
DOE/RL-96-0006	*4.2.2.1*	*Proven Engineering Practices/Margins-Proven Engineering Practices*
DOE/RL-96-0006	*4.2.2.3*	*Proven Engineering Practices/Margins-Safety System Design and Qualification*
DOE/RL-96-0006	*4.2.5.1*	*Inherent/Passive Safety Characteristics-Safety Margin Enhancement*

Safety Criterion: 4.1 - 3

Applicable Project Phases - Design and Construction

SSCs designated as ITS shall be designed to withstand the effects of NPH[1] events such as earthquakes, wind, and floods without loss of capability to perform specified safety functions. This includes both the primary and support systems that must function for an NPH event such that the public, co-located worker, or facility worker exposure standards of Safety Criteria 2.0-1 or 2.0-2 are not exceeded. The design shall consider both direct and indirect NPH effects, including common cause effects and interactions from failures of other SSCs. NPH design requirements for the various subcategories of ITS SSCs are described below.

The equivalence of the WTP Seismic Category to the seismic Performance Category of DOE-STD-1020-94 is as follows:

- Seismic Category-I is equivalent to Performance Category-3, except that the inelastic energy absorption factor shall be assumed to be 1.0.
- Seismic Category-II is equivalent to Performance Category-3, Seismic Category-III is equivalent to Performance Category-2, and Seismic Category-IV is equivalent to Performance Category-1.

<u>NPH Categorization of SSCs with SDC/SDS/RRC Safety Classification Scheme</u>:

1 For SDC SSCs that have an NPH safety function, the NPH design shall be as follows:
   a If the SSC has a seismic NPH safety function, the SSC shall be designated Seismic Category-I and designed to the seismic loadings provided in Table 4-1.
   b If the SSC has a non-seismic NPH safety function, the SSC shall be designated Performance Category-3 and designed to the corresponding non-seismic NPH loadings provided in Table 4-1.
2 For SDS SSCs whose failure under NPH conditions could adversely affect the NPH safety function(s) of an SDC SSC, the NPH design shall be as follows:
   a If the SSC failure from a seismic event could adversely affect the seismic NPH safety function(s) of an SDC SSC, the SSC shall be designated Seismic Category-II and designed to the seismic loadings provided in Table 4-1. (Note: for Seismic Category-II SSCs under this category, credit may be taken for inelastic energy absorption for seismic response.)
   b If the SSC failure from a non-seismic NPH event could adversely affect the non-seismic NPH safety function(s) of an SDC SSC, the SSC shall be designated Performance Category-3 and designed to the corresponding non-seismic NPH loadings provided in Table 4-1.
3 SDC SSCs that do not have an NPH safety function, SDS SSCs that do not adversely affect the NPH function of an SDC SSC, and RRC SSCs that provide primary confinement of significant inventories

[1] An SSC shall be considered to have a "NPH Safety Function" if its failure under NPH loads would result in unmitigated consequences greater than safety class Evaluation Guidelines.

of radioactive materials but in amounts less than quantities that require an SDC or SDS designation shall be:

a designed to the corresponding NPH loadings provided in Table 4-2

b designated Seismic Category-III

c designated non-seismic NPH Performance Category-2

4 RRC SSCs that do not provide primary confinement of significant inventories of radioactive materials shall be:

a designed to the corresponding NPH loadings provided in Table 4-2

b designated Seismic Category-IV

c designated Non- Seismic NPH Performance Category-1

NPH Categorization of SSCs with SC/SS/APC Safety Classification Scheme:[2]

1 NPH categories shall be assigned to SC/SS/APC SSCs as follows (assuming no SSC interaction effects exist):

a SC SSCs with a seismic safety function shall be assigned to Seismic Category (SC)-I and designed to seismic loadings provided in Table 4-1. SC SSCs with other NPH safety functions shall be assigned to Performance Category (PC)-3 and designed to the corresponding non-seismic loadings provided in Table 4-1.

b SS SSCs with a seismic safety function shall be assigned to SC-III and designed to the seismic loadings provided in Table 4-2. SS SSCs with other NPH safety functions shall be assigned to PC-2 and designed to the corresponding non-seismic NPH loadings provided in Table 4-2.

c SC and SS SSCs with no NPH safety functions shall be assigned to SC-III for seismic design and PC-2 for other non-seismic NPH design. These SC and SS SSCs shall be designed to corresponding NPH loadings provided in Table 4-2.

d APC SSCs (except those identified under interaction effects in Item 2 below) shall be assigned to SC-IV for seismic design and PC-1 for other non-seismic NPH design. These APC SSCs are designed to the corresponding NPH loadings provided in Table 4-2.

e An SSC shall be designated SC-III for seismic events and PC-2 for non-seismic events if its failure or in combination with one or more SSCs may result in loss of function of any emergency handling, hazard recovery, fire suppression, emergency preparedness, communication, or power system that may be needed to preserve the health and safety of workers and visitors.[3]

2 NPH categories shall be assigned to SC/SS/APC SSCs as follows, assuming SSC interaction effects (two over one protection):

a SSCs whose failure under seismic loads could prevent an SC SSC with a seismic safety function from performing that function shall be categorized as SC-II and designed to the seismic loadings provided in Table 4-1. Credit may be taken for inelastic energy absorption for seismic response in the design of SSCs under this category. SSCs whose failure under other NPH loads could prevent an SC SSC with a non-seismic NPH safety function from performing that function shall

[2] Requirements are in accordance with the guidance given in Sections 2.4 and 2.5 of DOE-STD-1021-93. For interaction effects, the "Potential for Interaction" in Table 2-1 of the Standard is taken as "High" (conservative).

[3] Requirement 1.e is to apply only until safe state has been achieved.

be categorized as PC-3 for the corresponding non-seismic NPH events and designed to the applicable non-seismic NPH loadings provided in Table 4-1.

b SSCs whose failure under seismic loads could prevent a SS SSCs with a seismic safety function from performing that function shall be categorized as SC-III and designed to seismic loadings provided in Table 4-2. SSCs whose failure under non-seismic NPH loads could prevent a SS SSC with a non-seismic NPH safety function from performing that function shall be categorized as PC-2 for the applicable non-seismic NPH events and designed to the corresponding non-seismic NPH loadings provided in Table 4-2.

c SSCs whose failure under seismic loads could prevent an APC SSC with a seismic function from performing that function shall be categorized as SC-IV and designed to seismic loadings provided in Table 4-2. SSCs whose failure under non-seismic NPH loads could prevent an APC SSC with a non-seismic NPH safety function from performing that function shall be categorized as PC-1 for the applicable non-seismic NPH events and designed to the corresponding non-seismic NPH loadings provided in Table 4-2.

Table 4-1 Natural Phenomena Design Loads Applicable to the NPH Safety Functions of SSCs that are Categorized as Seismic Category-I, Seismic Category-II, and Performance Category-3

Hazard	Load	Source Document for Load
Seismic	DBE with 0.26 *g* horizontal PGA and 0.18 *g* vertical PGA See Figures 4-1 and 4-2	WHC-SD-W236A-TI-002[a] DOE-STD-1020-94[b]
Straight wind	111 mi/hr, 3-second gust, at 33 ft above ground, Importance factor, I=1.0	DOE Newsletter [c]
Wind Missile	2×4 timber plank, 15 lb at 50 mi/hr (horiz), Max height 30 ft	DOE-STD-1020-94 [b]
Tornado and Tornado Missiles	Not Applicable	DOE-STD-1020-94 [b]
Volcanic ash	12.5 lb/ft^2	HNF-SD-GN-ER-501 [d]
Flooding	Dry site for river flooding Local precipitation: 4 in. for 6 hours	HNF-SD-GN-ER-501 [d]
Snow	15.0 lb/ft^2 snow load	HNF-SD-GN-ER-501 [d]

[a] Geomatrix, 1996, *Probabilistic Seismic Hazard Analysis DOE Hanford Site, Washington,* WHC-SD-W236A-TI-002, Rev.1A, prepared for Westinghouse Hanford Company, Richland, Washington.

[b] DOE STD-1020-94, (1996, Change 1) *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C., 1996.

[c] DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98.

[d] HNF-SD-GN-ER-501, Rev. 1, "Natural Phenomena Hazards, Hanford Site, South-Central Washington", Westinghouse Hanford Company.

Table 4-2. Natural Phenomena Design Loads Applicable to the NPH Design of ITS SSCs Where Table 4-1 Does Not Apply

Hazard	Load	Source Document for Load
Seismic	DOE-STD-1020-94 (Capacity from Uniform Building Code [a], Static Force Procedure)	DOE-STD-1020-94 [b]
Straight wind	91 mi/hr 3-second gust, at 33 ft above ground, Importance factor, I=1.00	DOE Newsletter [c]
Wind Missile	Not Applicable	DOE-STD-1020-94 [b]
Tornado and Tornado Missiles	Not Applicable	DOE-STD-1020-94 [b]
Volcanic ash	5 lb/ft^2	HNF-SD-GN-ER-501 [d]
Flooding	Dry site for river flooding Local Precipitation: 2.5 in. for 6 hours	HNF-SD-GN-ER-501 [d]
Snow	15.0 lb/ft^2 snow load	HNF-SD-GN-ER-501 [d]

[a] 1997, *Uniform Building Code,* International Conference of Building Officials, Whittier, California.

[b] DOE STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C., Change 1, 1996.

[c] DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98

[d] HNF-SD-GN-ER-501, Rev. 1, "Natural Phenomena Hazards, Hanford Site, South-Central Washington", Westinghouse Hanford Company

Figure 4-1

RPP-WTP DBE Horizontal Response Spectra
(0.50%, 2%, 3%, 4%, 5%, 7%, 10%, 12% Damping)

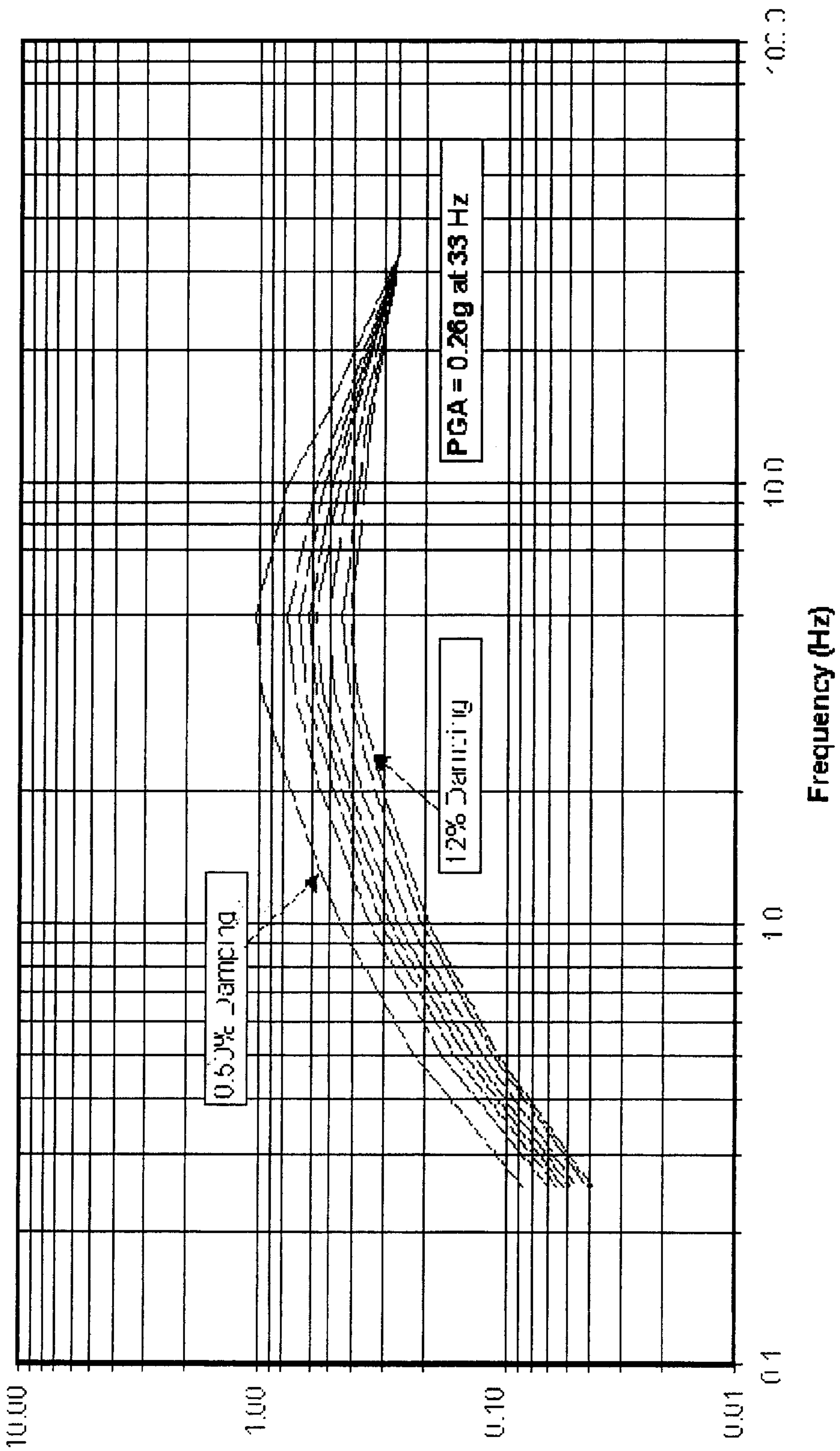
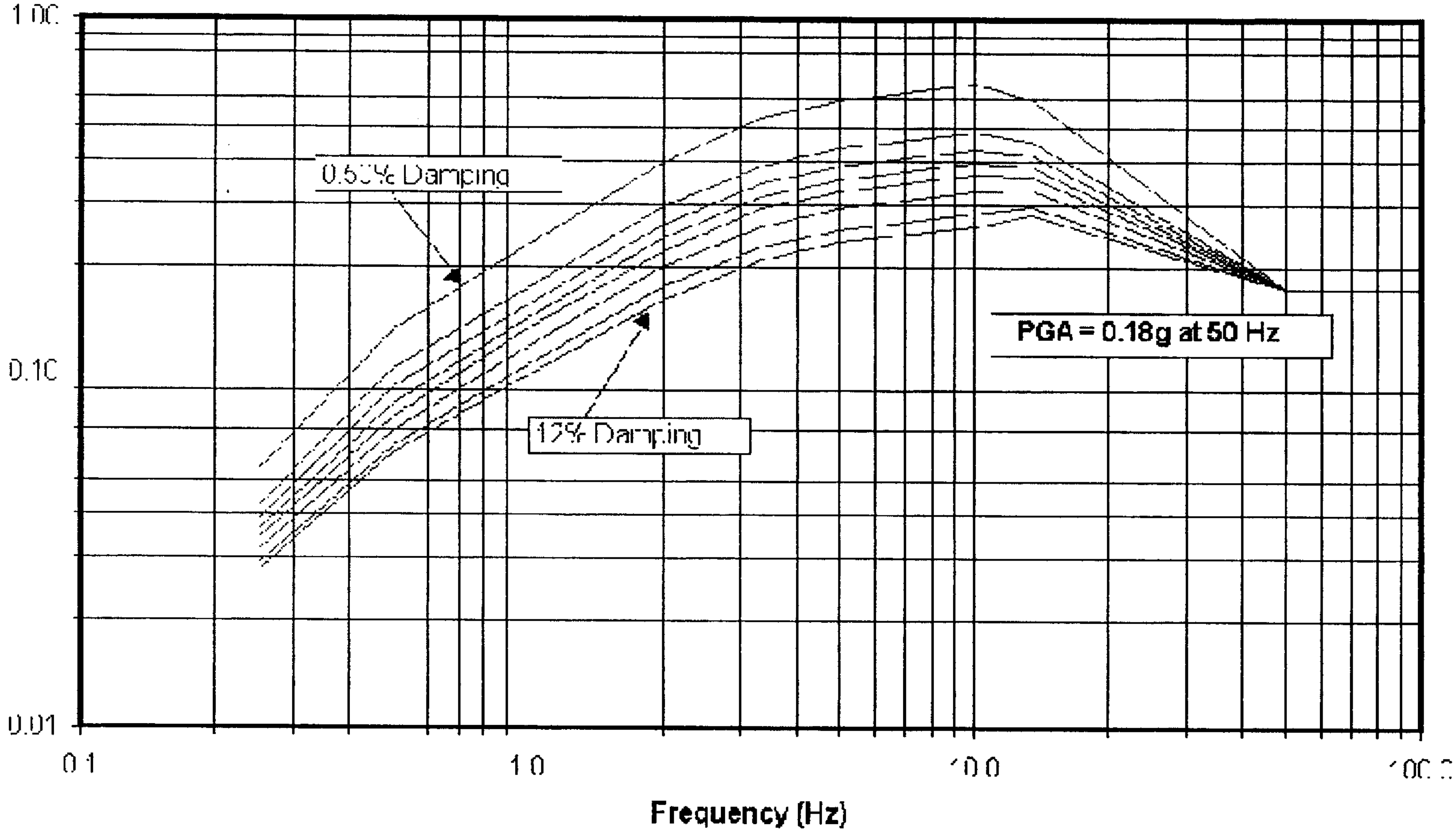


Figure 4-2

RPP-WTP DBE Vertical Response Spectra
(0.50%, 2%, 3%, 4%, 5%, 7%, 10%, 12% Damping)



Implementing Codes and Standards

ACI 318-99, *Building Code Requirements for Structural Concrete,* as tailored in Appendix C.
ACI 318R-99, *Commentary on Building Code Requirements for Structural Concrete*
ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures,* as tailored in Appendix C.
ACI 349R-01, *Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures*
ACI 530-99, *Building Code Requirements for Masonry Structures and Commentary*
ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities,* as tailored in Appendix C.
ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary*
ASCE 7-98, *Minimum Design Loads for Buildings and Other Structures*
AISC M016-89, *Manual for Steel Construction - Allowable Stress Design, Ninth Edition,* as tailored in Appendix C.
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
IEEE 344-1987 (R1993), *Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations,* as tailored in Appendix C.
IEEE 382-1996, *Standard for Qualification of Actuators for Power-Operated Valve Assemblies With Safety-Related Functions for Nuclear Power Plants*, as tailored in Appendix C.
1997, *UBC Uniform Building Code,* as tailored in Appendix C.
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006 *4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure*

Safety Criterion: 4.1 - 4

Applicable Project Phases - Design and Construction

Structures, systems, and components designated as Safety Design Class shall be appropriately protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from failures of moderate and high energy systems or other accident conditions.

In consideration of the need to protect structures, systems, and components which are designated as Safety Design Class from these dynamic effects, the failure of the moderate or high energy system need not be postulated to occur simultaneously with an accident unless the events are causally related.

Implementing Codes and Standards

ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures,* as tailored in Appendix C.
ACI 349R-01, *Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures*
ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities,* as tailored in Appendix C.
ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary*
ASCE 7-98, *Minimum Design Loads for Buildings and Other Structures*
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98

Safety Criterion: 4.1 - 5

Applicable Project Phases - All

Adequate provisions for facility security and physical protection of structures, systems, and components Important to Safety shall be provided. Safeguards and security provisions will be outlined in the Hanford Tank Waste Treatment Immobilization Safeguards and Security Plan. The plan will include the following topical elements:

- Program management
- Physical security
- Information security
- Computer security
- Personnel security

Implementing Codes and Standards

PL-W375-MG0004, *Safeguards and Security Program Plan*

Regulatory Basis

DOE/RL-96-0006 *4.3.6.1 Security-Security*

4.2 Confinement Design

Safety Criterion: 4.2 - 1

Applicable Project Phases - Design and Construction

The facility shall be designed to retain the radioactive and hazardous material through a conservatively designed confinement system for normal operations, anticipated operational occurrences, and accident conditions. The confinement system shall protect the worker and public from undue risk of releases such that the radiological and chemical exposure standards of Safety Criteria 2.0-1 and/or 2.0-2 are not exceeded.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix B, "Implementing Standard for Defense in Depth"
DOE G 420.1-1, *Nuclear Safety Design Criteria and Explosive Safety Criteria Guide for Use with DOE O 420.1, Facility Safety*
Section 2.3
DOE Order 420.1A, *Facility Safety*
Section 4.1.1.2

Regulatory Basis

DOE/RL-96-0006 *4.1.1.4 Defense in Depth-Mitigation*

Safety Criterion: 4.2 - 2

Applicable Project Phases - Design and Construction

Important to Safety liquid and gaseous systems and components, including pressure vessels, tanks, pumps, heat exchangers, piping, and valves, shall be designed to retain their hazardous inventory such that the radiological and chemical worker or public exposure standards of Safety Criteria 2.0-1 and/or 2.0-2 are not exceeded.

Implementing Codes and Standards

ASME B31.3-96, *Process Piping*, as tailored in Appendix C.
ASME SEC VIII, "Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels"
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
API STD 610-1995 Eighth Edition, "Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services", as tailored in Appendix C.
API STD 685-2000 First Edition, "Sealless Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services", as tailored in Appendix C.

Safety Criterion: 4.2 - 3

Applicable Project Phases - Design, Construction, and Operations (including Hot Commissioning)

Codes and standards for Important to Safety vessels and piping should be supplemented by additional measures (such as erosion/corrosion programs, piping in-service inspections, and seismic design and analysis) to mitigate conditions arising that could lead to a release of radiological or chemical material. The following are the additional measures for erosion/corrosion and piping in-service inspection to be considered in the material selection and vessel and piping design process:

1 Corrosion mechanisms such as general corrosion, pitting corrosion, end grain corrosion, stress corrosion cracking, crevice corrosion, corrosion at welds, microbiologically induced corrosion, fatigue corrosion, vapor phase corrosion, and galvanic corrosion.
2 Velocities above about 10 fps for slurries shall be evaluated for erosion.
3 High temperature vessels and piping shall be designed to allow for creep over the life of the component.
4 Where corrosion rates are closely predictable, corrosion allowance at least equal to the expected corrosion loss over a 40 year design life shall be specified.
5 Where corrosion rates are known, corrosion allowance, which includes any uncertainty in the corrosion rate, shall be specified.
6 Where the corrosion rates are indeterminate but expected to be low, a minimum standard corrosion allowance shall be specified.
7 When corrosion effects can be shown to be negligible or entirely absent, no corrosion allowance need be specified.
8 Where the solids content is greater than 2 % by weight, a minimum corrosion/erosion allowance shall be provided or hard overlay shall be provided in areas of high velocity.
9 An in-service inspection description as to where baseline measurements of welds or wall thicknesses should be taken and shall be made six months prior to hot commissioning to provide information that can be used to create an in-service inspections plan.
10 Vessels and piping with higher potential for corrosion or erosion shall be inspected within seven years after hot commissioning. Other vessels and piping with a lower potential for corrosion and erosion shall be inspected within ten years after hot commissioning.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*

Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix E, "Reliability, Availability, Maintainability, and Inspectability (RAMI)"
Appendix H, "Ad Hoc Implementing Standard for Erosion/Corrosion and Assessments"
Appendix L, "Ad Hoc Implementing Standard for Seismic Design of Pressure Vessels"

Regulatory Basis

DOE/RL-96-0006 *4.2.2.4 Proven Engineering Practices/Margins-Codes and Standards*

4.3 Engineered Safety Systems

Safety Criterion: 4.3 - 1

Applicable Project Phases - Design and Construction

Engineered safety systems shall be designed (1) to initiate automatically the operation of appropriate systems to assure that specified acceptable design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of Important to Safety systems and components. The ability to manually initiate engineered safety systems shall be provided.

Implementing Codes and Standards

ANSI/ANS 58.8-1994, *Time Response Design Criteria for Safety-Related Operator Actions*
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix B, "Implementing Standard for Defense in Depth"
ISA S84.01-96, *Application of Safety Instrumented Systems for the Process Industries*

Regulatory Basis

DOE/RL-96-0006 *4.1.1.5 Defense in Depth-Automatic Systems*

Safety Criterion: 4.3 - 2

Applicable Project Phases - Design and Construction

When single failure protection is required, Important to Safety engineered safety systems shall be designed to assure that the effects of natural phenomena (including lightning), and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix B, "Implementing Standard for Defense in Depth"
ANSI/ANS 58.9-1981, *Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems*
IEEE 323-83, *Qualifying Class 1E Equipment for Nuclear Power Generating Stations*, as tailored in Appendix C
IEEE 344-1987 (R1993), *Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*, as tailored in Appendix C
IEEE 379-1994, *Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems*, as tailored in Appendix C
IEEE 384-1992, *Standard Criteria for Independence of Class 1E Equipment and Circuits*, as tailored in Appendix C
NFPA 780-97, *Standard for the Installation of Lightning Protection Systems*
NFPA 801-2003, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, as tailored in Appendix C

Safety Criterion: 4.3 - 3

Applicable Project Phases - Design and Construction

Important to Safety engineered safety systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Design provisions should be included to limit the loss of safety functions due to damage to several structures, systems, or components Important to Safety resulting from a common-cause or common-mode failure.

The protection system shall be designed to permit periodic testing of its functioning when the facility is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Implementing Codes and Standards

IEEE 338-1987, *Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems*, as tailored in Appendix C

IEEE 379-1994, *Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems*, as tailored in Appendix C

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*

Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006 *4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure*

Safety Criterion: 4.3 - 4

Applicable Project Phases - Design and Construction

Important to Safety instrumentation and controls shall be provided to monitor variables and systems and control systems and components over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate public and worker safety by compliance to the standards of Safety Criteria 2.0-1 and 2.0-2, including those variables and systems that can affect the performance of Important to Safety facility conditions. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges. The instrumentation and controls provided shall provide the ability to detect off normal conditions, mitigate accidents, and place the facility in a safe state.

Implementing Codes and Standards

ANSI N42.18-1980 (R 1991), Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents

DOE G 420.1-1, Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria Guide for use with DOE O 420.1, Facility Safety, Section 2.3

DOE Order 420.1A, Facility Safety, Section 4.1.1.2

IEEE-497-2002, *Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations* as tailored in Appendix C

ISA S12.13 PT 1-95, Performance Requirements, Combustible Gas Detectors

ISA S84.01-96, Application of Safety Instrumented Systems for the Process Industries

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II

Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Appendix B, "Implementing Standard for Defense in Depth"

Regulatory Basis

DOE/RL-96-0006	*4.1.1.3*	*Defense in Depth-Control*
DOE/RL-96-0006	*4.2.6.2*	*Human Factors-Instrumentation and Control Design*

Safety Criterion: 4.3 - 5

Applicable Project Phases - Design and Construction

When single failure protection is required, Important to Safety protection systems shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix B, "Implementing Standard for Defense in Depth"

IEEE 384-1992, *Standard Criteria for Independence of Class 1E Equipment and Circuits*, as tailored in Appendix C

Safety Criterion: 4.3 - 6

Applicable Project Phases - Design and Construction

The possibility of human error in facility operations shall be taken into account in the design by facilitating correct decisions by operators and inhibiting wrong decisions and by providing means for detecting and correcting or compensating for error. The parameters to be monitored in control areas shall be selected and their displays arranged to ensure operators have clear and unambiguous indication of the status of the facility. The parameters and displays shall facilitate monitoring and the initiation and operation of systems designated as Important to Safety.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix B, "Implementing Standard for Defense in Depth"

IEEE 1023-88, *Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations*

Regulatory Basis

DOE/RL-96-0006	*4.1.1.6*	*Defense in Depth-Human Aspects*
DOE/RL-96-0006	*4.2.6.1*	*Human Factors-Human Error*
DOE/RL-96-0006	*4.2.6.2*	*Human Factors-Instrumentation and Control Design*
DOE/RL-96-0006	*4.2.6.3*	*Human Factors-Safety Status*

Safety Criterion: 4.3 - 7

Applicable Project Phases - Design and Construction

The control room or control area shall be designed to permit occupancy and actions to be taken to monitor the facility safely during normal operations, and to provide safe control of the facility for anticipated operational occurrences and accident conditions. If credit is taken for operator action to satisfy the accident exposure standards of Safety Criteria 2.0-1 and/or 2.0-2, adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE), 30 rem thyroid, and 30 rem beta skin for the duration of the accident. In the event operator action is not required, other than immediate actions required to place the facility operation into a safe state, then the worker exposure standards of Safety Criterion 2.0-1 apply. For occurrences and accidents involving chemical release, provisions shall be made such that the operator exposure does not exceed the worker exposure standards of 29 CFR 1910.120 for emergency exposure.

Consideration shall also be given to accidents at nearby facilities if operator action is required to safely control the processes and bring them to a safe state.

The need for an alternate system that would allow the processes to be placed in a safe state in the event the primary control area is uninhabitable shall be evaluated.

Implementing Codes and Standards

ASME N509-89, *Nuclear Power Plant Air Cleaning Units and Components*
ASME N510-1989 (R 1995), *Testing of Nuclear Air Cleaning Systems*
NUREG-0800, *Standard Review Plan,* Section 6.4, Section II, Items 1-6, Draft Rev. 3, April 1996, as tailored in Appendix C.
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006	*4.2.4.1 Emergency Preparedness-Support Facilities*
DOE/RL-96-0006	*4.2.6.2 Human Factors-Instrumentation and Control Design*
29 CFR 1910.120	*Hazardous Waste Operations and Emergency Response*

4.4 Electrical and Mechanical Systems

Safety Criterion: 4.4 - 1

Applicable Project Phases - Design and Construction

Structures, systems, and components Important to Safety shall be designed and qualified to function as intended in the environments associated with the events for which they are intended to respond. The effects of aging on normal and abnormal functioning shall be considered in design and qualification.

Implementing Codes and Standards

10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power"
IEEE 323-83, *Qualifying Class 1E Equipment for Nuclear Power Generating Stations,* as tailored in Appendix C
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

Regulatory Basis

DOE/RL-96-0006	*4.2.2.3 Proven Engineering Practices/Margins-Safety System Design and Qualification*
10 CFR 50.49	*Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power*

Safety Criterion: 4.4 - 2

Applicable Project Phases - Design and Construction

Structures, systems, and components Important to Safety shall be designated, designed and constructed to permit appropriate inspection, testing, and maintenance throughout their operating lives to verify their continued acceptability for service with an adequate safety margin.

Systems and components designated as Important to Safety that are located in closed cells where access is not possible during facility operation or scheduled shutdown periods shall be designed and constructed to standards aimed at ensuring their suitability for the entire service life with an adequate safety margin. Alternately, provisions may be made for remote replacement, standby cells, or equipment or other methods capable of ensuring a serviceable facility with adequate safety for the duration of the intended operating life.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix E, "Reliability, Availability, Maintainability, and Inspectability (RAMI)"
IEEE 338-1987, *Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,* as tailored in Appendix C
ISA S84.01-1996, *Application of Safety Instrumented Systems for the Process Industries*

Regulatory Basis

DOE/RL-96-0006	*4.2.7.1 Reliability, Availability, Maintainability, and Inspectability (RAMI)-Reliability*
DOE/RL-96-0006	*4.2.7.2 Reliability, Availability, Maintainability, and Inspectability (RAMI)-Availability, Maintainability, and Inspectability*

Safety Criterion: 4.4 - 3

Applicable Project Phases - Design and Construction

Ventilation systems and off-gas systems must be provided where necessary to control radiological and chemical material releases and the generation of flammable and explosive gases during normal and accident conditions. The design shall permit appropriate periodic inspection and testing; SDC air treatment systems shall have suitable redundancy to ensure its safety function can be accomplished, assuming a single failure. SDS air treatment systems shall be designed to ensure their operability under normal conditions. SS air treatment systems shall be designed to ensure their operability under normal and accident conditions and the single failure criteria shall be considered for active components in the system.

Implementing Codes and Standards

ASME AG-1-1997 with ASME AG-1a-2000 Addenda, *Code on Nuclear Air and Gas Treatment*
ASME N509-89, *Nuclear Power Plant Air Cleaning Units and Components*
ASME N510-1989 (R 1995), *Testing of Nuclear Air Cleaning Systems*
ASME B31.3-1996, *Process Piping*, as tailored in Appendix C
ASME SEC VIII, Boiler and Pressure Vessel Codes, *Rules for Construction of Pressure Vessels*
NFPA 801-03, *Standard for Facilities Handling Radioactive Materials*, as tailored in Appendix C

Regulatory Basis

DOE-RL-96.0006 *4.2.1.1 Design - Safety Design*

Safety Criterion: 4.4 - 4

Applicable Project Phases - Design and Construction

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of systems designated as Safety Design Class. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure Safety Design Class functions are maintained in the event of postulated accidents. Onsite electric power systems shall be provided to permit functioning of SDC systems that require electrical power to perform their safety functions during loss of offsite power as determined by the accident analysis. The onsite power systems shall include sufficient independence, redundancy, and testability to ensure that the safety function can be performed under postulated accident conditions, including a single failure if postulated. Physical and electrical separation shall be provided between diverse or redundant SDC electrical systems.

Implementing Codes and Standards

IEEE 308-91, *Criteria for Class 1E Power Systems for Nuclear Power Generating Stations*, as tailored in Appendix C.
IEEE 338-1987, *Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems*, as tailored in Appendix C
IEEE 344-1987 (R 1993), *Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*, as tailored in Appendix C
IEEE 384-1992, *Standard Criteria for Independence of Class 1E Equipment and Circuits*, as tailored in Appendix C
IEEE 387-1995, *Standard Criteria for Diesel-Generator Units Applied as Standby Power Generating Stations*, as tailored in Appendix C.
IEEE 450-1995, *Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations*

IEEE 484-1996, *Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations*
IEEE 485-1983, *Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations*
IEEE 628-1987, *Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations,* as tailored in Appendix C
IEEE 741-1990, *Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations,* as tailored in Appendix C
IEEE 946-1992, *Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations*

Safety Criterion: 4.4 - 5

Applicable Project Phases - Design and Construction

Electric power systems designated as Safety Design Significant shall be designed to ensure their operability under normal conditions. Electric power systems designated as Safety Significant shall be designed to ensure their operability under normal and accident conditions and the single failure criteria shall be considered for active components in the system.

The design shall permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to periodically test:

(1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses

(2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system

Implementing Codes and Standards

IEEE 338-1987, *Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,* as tailored in Appendix C
IEEE 384-1992, *Standard Criteria for Independence of Class 1E Equipment and Circuits,* as tailored in Appendix C
NFPA 70-1999, *National Electric Code*

Safety Criterion: 4.4 - 6

Applicable Project Phases - Design and Construction

Air to Important to Safety instrumentation and valve actuators (regardless of whether air operation of the valve actuators is Important to Safety) shall provide clean, dry, and oil free air, and shall be free of all corrosive and hazardous gases.

Implementing Codes and Standards

ISA S7.0.01-1996, *Quality Standard for Instrument Air*
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II,* Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"

4.5 Fire Protection

Safety Criterion: 4.5 – 1

Applicable Project Phases - All

WTP facilities, sites and activities (including design and construction) shall be characterized by a level of fire protection that is sufficient to fulfill the requirements of the best protected class of industrial risks ("Highly Protected Risk" or "Improved Risk") and shall be provided protection to achieve defense-in-depth.

The fire protection design features for WTP facilities shall be developed, implemented, and maintained that includes the design requirements of DOE O 420.1A and two reliable and separate water supplies of adequate capacity for fire protection. Redundant Safety Design Class systems (for the protection of the worker and co-located worker) should be in separate fire areas.

Implementing Codes and Standards

DOE O 420.1A, *Facility Safety,* as tailored in Appendix C
DOE-STD-1066-97, *Fire Protection Design Criteria*
NFPA 801-2003, *Standard for Fire Protection for Facilities Handling Radioactive Materials,* as tailored in Appendix C

Regulatory Basis

10 CFR 830 Subpart B *Safety Basis Requirements*

Safety Criterion: 4.5 – 2

Applicable Project Phases - All

A fire protection program shall be developed, implemented, and maintained that will minimize the potential for:

(1) The occurrence of a fire or related event
(2) A fire that causes an on-site or off-site release of hazardous materials that exceeds SRD Safety Criterion 2.0-2
(3) A fire that causes an on-site or off-site release of radioactive materials that exceed SRD Table 2-1, "Radiological Exposure Standards Above Normal Background"
(4) Property losses from a fire and related events exceeding defined limits established by DOE.

The fire protection program will also:

(1) Limit the damage to Safety Design Class systems (for the protection of the public only) as a result of a fire and related events
(2) Include surveillance to ensure that fire barriers are in place and that fire suppression systems and components are operable; and
(3) Designate staff members responsible for fire protection review of proposed work activities

The Fire Protection Program shall include the general programmatic requirement of DOE O 420.1A.

Implementing Codes and Standards

DOE O 420.1A, *Facility Safety,* as tailored in Appendix C
DOE-STD-1066-97, *Fire Protection Design Criteria*

NFPA 801-2003, *Standard for Fire Protection for Facilities Handling Radioactive Materials,* as tailored in Appendix C

Regulatory Basis
10 CFR 830 Subpart B *Safety Basis Requirements*

Safety Criterion: 4.5 – 3

Applicable Project Phases - All

A Fire Hazard Analysis (FHA) shall be performed for all nuclear facilities, significant new facilities, and facilities that represent unique or significant fire safety risks. Such a systematic analysis shall divide the facility into "fire areas" and evaluate the fire safety of each area and of the facility as a whole. The conclusions of the FHA shall be incorporated in the Safety Analysis Report (SAR) and shall be integrated into design basis and beyond design basis accident conditions. The analysis shall, for each fire area:

1. Account for all radioactive, hazardous, and combustible materials, including estimates of their heat content
2. Describe the processes performed and their potential for fire or explosion
3. Account for the sources of heat and flame
4. List the fire detection and suppression equipment
5. Consider credible fire scenarios and evaluate the adequacy of the fire protection measures
6. Document Maximum Possible Fire Loss (MPFL) for all nuclear facilities, significant new facilities, and facilities that represent unique or significant fire safety risks.

In addition, the FHA shall consider other buildings or installations close to process buildings that contain flammable, combustible, or reactive liquid or gas storage.

The FHA shall confirm that the facility can be placed in a safe state during and after all credible fire and explosion conditions.

Implementing Codes and Standards
DOE O 420.1A, *Facility Safety,* as tailored in Appendix C
DOE-STD-1066-97, *Fire Protection Design Criteria*
NFPA 801-2003, *Standard for Fire Protection for Facilities Handling Radioactive Materials,* as tailored in Appendix C

Regulatory Basis
10 CFR 830 Subpart B Safety Basis Requirements

Safety Criterion: 4.5 – 4

Applicable Project Phases - All

Hot work permits shall be issued for hot work operations conducted in or near the facility. The permit shall document that applicable fire prevention and protection requirements have been implemented prior to beginning the hot work operations; it shall indicate the date(s) authorized for hot work; and identify the object on which hot work is to be performed. The permit shall be kept on file until completion of the hot work operations.

Implementing Codes and Standards

DOE O 420.1A, *Facility Safety*, as tailored in Appendix C

NFPA 801-2003, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006 *5.2.8 Hot Work Control*

5.0 Radiation Protection

Safety Criterion: 5.0 - 1

Applicable Project Phases - All

A Radiation Protection Program (RPP) compliant with 10 CFR 835 and DOE O 420.1A shall be developed and submitted for approval to DOE.

The WTP Radiological Controls Program shall address all items in 10 CFR 835 and the additional Safety Criteria provided in SRD Volume II Section 5.0.

Implementing Codes and Standards

DOE O 420.1A, *Facility Safety*
DOE G 441.1-1, *Management and Administration of Radiation Protection Programs Guide*

Regulatory Basis

10 CFR 835 Occupational Radiation Protection Location: 101(a-f)

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*3.3.2*	*Worker Protection*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation Protection Practices*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Radiation Features*
DOE/RL-96-0006	*4.3.2.1*	*Radiation Protection-Radiation Practices*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

Safety Criterion: 5.0 - 2

Applicable Project Phases - All

A respiratory protection program shall be established that includes:

(1) Use of respiratory protection equipment, including equipment used as emergency devices, that is tested and certified or had certification extended by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA).

(2) Air sampling sufficient to identify the potential hazard, permit proper equipment selection, and estimate exposures.

(3) Surveys and bioassays, as appropriate, to evaluate actual intakes.

(4) Testing of respirators for operability immediately prior to each use.

(5) Written procedures regarding selection, fitting, issuance, maintenance, and testing of respirators, including testing for operability immediately prior to each use; supervision and training of personnel; monitoring, including air sampling and bioassays; and recordkeeping.

(6) Determination by a physician prior to the initial fitting of respirators, and either every 12 months thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

(7) A written policy statement on respirator usage covering:

(i) The use of process or other engineering controls, instead of respirators.

(ii) The routine, nonroutine, and emergency use of respirators.

(iii) The periods of respirator use and relief from respirator use. Each respirator user will be informed that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.

(8) Use of equipment within limitations for type and mode of use and provision for proper visual, communication, and other special capabilities (such as adequate skin protection) when needed.

(9) Notification to the Regulator, in writing, at least 30 days before the date that respiratory protection equipment is first used to protect workers from airborne radioactivity.

Implementing Codes and Standards
ANSI Z-88.2-1992, *American National Standard for Respiratory Protection*

Regulatory Basis
29 CFR 1910.134, *Respiratory Protection*

5.1 Environmental Radiation Protection

Safety Criterion: 5.1 - 1

Applicable Project Phases - All

An Environmental Radiological Protection Program shall be prepared and submitted to the regulator.

The Environmental Radiological Protection Program (ERPP) shall address the following elements, as appropriate:

(1) the identity of existing and anticipated types of activities and areas of the site subject to the ERPP

(2) the measures to be used to implement the ERPP

(3) the methods to be used to monitor, report, and record compliance with the ERPP

(4) models and methods used for dose assessment including bioaccumulation and dose-conversion factors

(5) an As Low As is Reasonably Achievable (ALARA) Program

(6) effluent and environmental monitoring including:

- (i) sources of airborne emissions
- (ii) sources of discharges in liquid waste streams
- (iii) effluent monitoring
- (iv) environmental surveillance
- (v) meteorological data acquisition
- (vi) pre-operational evaluation

(7) ground water protection

(8) radiological protection in the management of radioactive waste

(9) controls on the release of materials

(10) property containing residual radioactive materials

Implementing Codes and Standards

ANSI/ISO-14001-1996, *Environmental Management Systems - Specifications with guidance for use*

Regulatory Basis

DE-AC06-96RL13308	*Part I Section C.5 Table S4-1*	
DOE/RL-96-0006	*4.3.2.1*	*Radiation Protection-Radiation Practices*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

Safety Criterion: 5.1 - 2

Applicable Project Phases - All

Environmental emissions of radioactive effluents and doses to the public, including air and liquid effluents and wastes, shall be ALARA and compliant with prescribed limits, and ensure mitigation of the extent of radiation exposure and environmental impact due to accidents. Equipment shall be designed, installed, and operated to monitor and maintain control over radioactive materials in air and liquid effluents produced during normal operations and accidents. The system of radiation protection practices for design, installation, and operation of radioactive air and liquid effluent equipment, including monitoring systems, shall ensure environmental radiation and doses to the public are ALARA and in compliance with prescribed limits. Computer codes or procedures used to determine the total effective dose equivalent from environmental radiation emissions shall be EPA approved.

Implementing Codes and Standards

WAC 246-221[3/24/01] Radiation Protection Standards
WAC 246-247[7/9/98] Radiation Protection - Air Emissions
40 CFR 61, Subpart H National Emission Standards for Hazardous Air Pollutants
WAC 173-303[4/13/03] Dangerous Waste Regulations
WAC 173-216 [3/18/02] State Waste Discharge Permit Program
WAC 246-272 [1/1/95] On-Site Sewage Systems

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation Protection Practices*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Features*
DOE/RL-96-0006	*4.3.2.1*	*Radiation Protection-Radiation Practices*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

Safety Criterion: 5.1 - 3

Applicable Project Phases - All

A waste management program shall be developed and maintained. The waste management program shall ensure radiation emissions and doses to the general public and environment due to radioactive wastes arising from WTP operations and anticipated operational occurrences shall be ALARA and shall comply with prescribed limits. Measurements of environmental radiation doses to the public from radioactive and mixed waste shall be performed to demonstrate compliance with prescribed limits.

Controls on the release of materials and property containing residual radioactive material as a direct result of WTP Operations shall be established, shall be ALARA, and shall comply with prescribed limits. Monitoring equipment and systems used for release of materials and property shall demonstrate compliance with prescribed environmental radiation dose limits. Materials and equipment that have inaccessible areas or are potentially contaminated by volume shall not be released from radiological control. Written procedures shall be developed to control activities described in the above areas.

Implementing Codes and Standards

WAC 173-303 [4/13/03] Dangerous Waste Regulations
WAC 246-232-140 [12/29/00] Schedule D, Acceptable Surface Contamination Levels
WAC 246-246-020 [6/29/01] Radiological Criteria for Unrestricted Use

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation Protection Practices*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Radiation Protection Features*
DOE/RL-96-0006	*4.3.2.1*	*Radiation Protection-Radiation Practices*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

Safety Criterion: 5.1 - 4

Applicable Project Phases - All

A legacy radioactive materials program shall be developed and maintained for controlling the release of WTP materials and property from the Hanford Site. The monitoring program for legacy radioactive materials shall be described in the PSAR/SAR. Any identified radioactive material above background shall be posted, labeled, and packaged in accordance with the Radiological Control Program. All WTP releases from the Hanford Site shall be performed in accordance with the Radioactive Materials Management Program. Since the detection level of the monitoring program is not capable of detecting volumetric contamination, large quantities of soil or concrete (if made using Hanford Site soil) shall not be removed from the Hanford Site.

Implementing Codes and Standards

WAC 173-303 [4/13/03] Dangerous Waste Regulations
WAC 246-232-140 [12/29/00] Schedule D, Acceptable Surface Contamination Levels
WAC 246-246-020 [6/29/01] Radiological Criteria for Unrestricted Use

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation Protection Practices*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Radiation Protection Features*
DOE/RL-96-0006	*4.3.2.1*	*Radiation Protection-Radiation Practices*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

5.2 Environmental Radiological Monitoring

Safety Criterion: 5.2 - 1

Applicable Project Phases - All

Measurement of environmental radiation doses to the public shall be performed to demonstrate compliance with prescribed limits. A system of radiation protection practices for the design, installation, and operation of monitoring equipment and systems for air and liquid effluents, including non-point and fugitive emissions, shall be ALARA and in compliance with the prescribed limits. Environmental radiation effluent and dose measurements and calculations, records sufficient to demonstrate compliance with prescribed environmental radiation and dose limits and information sufficient for mandatory state and federal environmental radiation effluent and public dose reporting shall be prepared and maintained. Written procedures shall be developed to control activities described in the above areas.

Implementing Codes and Standards

WAC 246-221[3/24/01] Radiation Protection Standards
WAC 246-247 [7/9/98] Radiation Protection - Air Emissions
40 CFR 61, Subpart H National Emission Standards for Hazardous Air Pollutants
WAC 173-303[4/13/03] Dangerous Waste Regulations
WAC 173-216 [3/18/02] State Waste Discharge Permit Program
WAC 246-272 [1/1/95] On-Site Sewage Systems

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation ProtectionPractices*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Features*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Radiation Features*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

6.0 Startup

Safety Criterion: 6.0 – 1

Applicable Project Phases – Design and Construction, Cold Commissioning and Operations (including Hot Commissioning)

A testing program shall be established and followed to demonstrate that Important to Safety structures, systems, and components have been properly constructed and can perform their specified functions. The program shall provide for the detection, tracking, and correction of deficiencies. The testing program shall be developed using the graded approach and address the following elements:

1 Test Phase
2 Test Procedures
3 Validation of Operating and Maintenance Procedures
4 Test Acceptance Criteria
5 Correction of Deficiencies
6 Training and Qualification of Personnel
7 Retest
8 Readiness Assessment
9 Records

Implementing Codes and Standards
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document*
Appendix J, "Ad Hoc Implementing Standard for Startup"

Regulatory Basis
DOE/RL-96-0006 *4.2.8.1* *Pre-Operational Testing-Testing Program*

Safety Criterion: 6.0 – 2

Applicable Project Phases – Design and Construction, Cold Commissioning and Operations (including Hot Commissioning)

Procedures for normal facility and systems operation and for functional tests to be performed during the operating phase shall be validated as part of the component, system, and commissioning testing program. Operations procedures for the WTP will be drafted, reviewed, verified, validated, and approved per the WTP Conduct of Operations Program. Validated procedures will be provided to the testing organization for use during initial system startup and other testing activities as needed.

Implementing Codes and Standards
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document*
Appendix J, "Ad Hoc Implementing Standard for Startup"

Regulatory Basis
DOE/RL-96-0006 *4.2.8.2* *Pre-Operational Testing-Operational Systems and Functional Testing Procedures Validation*

Safety Criterion: 6.0 – 3

Applicable Project Phases – Design and Construction, Cold Commissioning and Operations (including Hot Commissioning)

During component, system, and commissioning testing, detailed diagnostic data shall be collected on systems and components designated as Important to Safety and the initial operating parameters of the systems and components shall be recorded.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document*
Appendix J, "Ad Hoc Implementing Standard for Startup"

Regulatory Basis

DOE/RL-96-0006 *4.2.8.3 Pre-Operational Testing-Safety Systems Data*

Safety Criterion: 6.0 – 4

Applicable Project Phases – Design and Construction, Cold Commissioning and Operations (including Hot Commissioning)

During component, system, and commissioning testing program, the as-built operating characteristics of process systems, and systems and components designated as Important to Safety shall be determined and documented. Operating points shall be adjusted to conform to values in the design basis. Training procedures and limiting conditions for operation shall be modified, if necessary, to accurately reflect the operating characteristics of the systems and components as built.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document*
Appendix J, "Ad Hoc Implementing Standard for Startup"

Regulatory Basis

DOE/RL-96-0006 *4.2.8.4 Pre-Operational Testing-Design Operating Characteristics*

Safety Criterion: 6.0 – 5

Applicable Project Phases – Cold Commissioning and Operations (including Hot Commissioning)

A pre-startup safety review shall be performed. The pre-startup safety review shall confirm that, prior to the introduction of radioactive or process chemicals considered to pose a hazard to a process, construction and equipment is in accordance with design specifications; safety, operating, maintenance, and emergency procedures are in place and are adequate; a process hazard analysis has been performed and recommendations have been resolved or implemented before startup; and training of each employee involved in operating a process has been completed.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document*
Appendix J, "Ad Hoc Implementing Standard for Startup"

Regulatory Basis

DOE/RL-96-0006	*4.3.1.4*	*Conduct of Operations-Readiness*
DOE/RL-96-0006	*5.2.6*	*Pre-startup Safety Review*

7.0 Management and Operations

7.1 Management and Organization/Staffing

Safety Criterion: 7.1 - 1

Applicable Project Phases - All

The Contractor shall conduct a compliance audit periodically to verify that the procedures and practices developed under the process safety management program are adequate and being followed. The frequency of compliance audits shall be based on the applicable standards and the nature of the process hazards. The Contractor shall promptly determine and document an appropriate response to each finding of the compliance audit. The results of the audits shall be available to the DOE in support of regulatory oversight.

Implementing Codes and Standards

24590-WTP-QAM-QA-001, *Quality Assurance Manual*
Policy Q-18.1, "Independent Assessment (Audit)"

Regulatory Basis

DOE/RL-96-0006 5.2.12 Compliance Audits

Safety Criterion: 7.1 - 2

Applicable Project Phases - All

Subcontractors may be utilized to perform a variety of work. Subcontractors past safety performance shall be evaluated prior to contract award. Subcontractors shall ensure that:

- Employees are trained in work practices necessary to safely perform their work
- Employees are instructed in known hazards of the process as related to their job assignments, and in relevant portions of the emergency management plan
- It is documented that each employee has received and understood the training required to work safely
- Employees follow safety rules of the WTP site safe work practices, and advise the contractor of any unique hazards presented or found during the course work

The WTP Project Contractor shall:

- Inform subcontractors of known fire, explosion, or toxic hazards related to the subcontractor work or process
- Explain applicable provisions of the emergency management plan
- Develop and implement safe work practices to control entrance, presence, and exit of subcontractor employees

- Periodically evaluate performance of subcontractors
- Maintain an illness and injury log relating to subcontractor

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II,* Appendix I - "Ad Hoc Implementing Standard for Project Integrated Safety Management Approach"

Regulatory Basis

DOE/RL-96-0006 *5.2.5* *Subcontractors*

Safety Criterion: 7.1 - 3

Applicable Project Phases - All

A framework shall be established for safety review organizations that are responsible for assuring the safety of the facility. The separation between the responsibilities of the safety review organizations and those of the other organizations shall remain clear so that the safety review organizations retain their independence as safety authorities. Internal safety oversight should be conducted by qualified personnel to ensure that the safety standards are consistently met. Internal safety oversight functions include corporate safety assessments, management assessments, continued surveillance, independent assessments and audits, safety committees, incident investigations, maintenance of the authorization basis, and, during radiological operations, the USQ process. The following activities are part of internal safety oversight:

1 Reviewing the design for safety consequences and consistency with regulatory requirements
2 Reporting deficient conditions to line management
3 Reviewing procedures, programs, plans, and management processes for consistency with regulatory requirements
4 Conducting safety oversight and management assessments
5 Assisting line management to establish a positive safety culture
6 Incorporating applicable lessons learned from previous WTP incidents and industry experience at other DOE sites and the commercial power industry relevant to the Project oversight program
7 Maintaining a continuing interaction with the WTP Project Regulator on the status and direction of safety oversight activities

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II,* Appendix I - "Ad Hoc Implementing Standard for Project Integrated Safety Management Approach"

Regulatory Basis

DOE/RL-96-0006	*4.1.4.1*	*Safety/Quality Culture-Safety/Quality Culture*
DOE/RL-96-0006	*4.3.1.5*	*Conduct of Operations-Internal Surveillance and Audits*
DOE/RL-96-0006	*4.4.1*	*Safety Review Organization*
DOE/RL-96-0006	*4.4.2*	*Qualified Personnel*
DOE/RL-96-0006	*5.1.3*	*Process Safety Responsibility*

Safety Criterion: 7.1 - 4

Applicable Project Phases - All

Commitments from outside organizations to provide data and services required to satisfy safety obligations shall be made prior to the need for the information or services.

Implementing Codes and Standards

DOE/RL-94-02, *Hanford Emergency Management Plan*, as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006 *4.1.2.3 Safety Responsibility-Site and Technical Support*

7.2 Training and Procedures

Safety Criterion: 7.2 – 1

Applicable Project Phases - All

Programs providing training and qualifications for operations, maintenance and technical support personnel to enable them to perform their duties safely and efficiently will be developed and implemented utilizing a graded approach. Training will be developed using the systematic approach to training (SAT) and include the requirements for the following:

1 Training organization;
2 Subcontractor personnel qualifications;
3 Personnel selection;
4 Qualification process;
5 Training and qualification;
6 Operator and supervisor examination;
7 Requalification; and
8 Alternatives to education and experience

Implementing Codes and Standards

DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities*, Attachment 2, "References and Definitions", as tailored in, Appendix C

Regulatory Basis

DOE/RL-96-0006	*4.3.1.7*	*Conduct of Operations-Access to Technical Safety Support*
DOE/RL-96-0006	*4.3.4.2*	*Training and Qualifications-Training Programs*

Safety Criterion: 7.2 – 2

Applicable Project Phases – Cold Commissioning and Operartions (Including Hot Commissioning)

Each employee involved in operating a process shall be trained in an overview of the process and in the operating procedures/instructions. The training shall include emphasis on the specific safety and health hazards, operating limits, emergency operations including shutdown, and safe work practices applicable to the employee's job tasks.

Refresher training shall be provided at least every three years, and more often if necessary, to each employee involved in operating a process to assure that the employee understands and adheres to the current operating procedures/instructions of the process and is proficient in the procedures to follow if conditions exceed the design basis of the facility.

Implementing Codes and Standards

DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities*,
Attachment 2, "References and Definitions", as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006 *4.3.4.1* *Training and Qualifications-Personnel Training*
DOE/RL-96-0006 *4.3.4.3* *Training and Qualifications-Conditions Beyond Design Basis*
DOE/RL-96-0006 *5.2.4* *Training*

Safety Criterion: 7.2 – 3

Applicable Project Phases - All

Written procedures/instructions that provide clear direction for safely conducting activities involving radioactive or hazardous materials shall be developed and implemented for each phase of the facility life. The procedures/instructions shall address at least the following elements:

1 Steps for each operating phase
2 Operating limits
3 Safety and health considerations
4 Safety systems and their functions

Project procedures will be prepared to provide explicit instruction for accomplishing work and to support management control function and technical work activities. Administrative procedures are used to implement management control functions, control the interactions among WTP project organizations, assist in ensuring that work is performed systematically and correctly. Procedures will be prepared during the appropriate phases of the project to support activities such as:

1 Configuration Management
2 Design
3 Construction
4 Testing
5 Startup
6 Operations
7 Periodic Surveillance
8 Maintenance
9 Emergency Preparedness
10 Fire Protection
11 Training and Qualifications
12 Work Planning
13 Quality Assurance
14 Management Assessments
15 Safeguards and Security

Implementing Codes and Standards

DOE Order 5480.19, *Conduct of Operations Requirements for DOE Facilities*, as tailored in Appendix C
DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities*, Attachment 2, "References and Definitions" as tailored in Appendix C.

Regulatory Basis

DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*
DOE/RL-96-0006	*5.2.3*	*Operating Procedures*

7.3 Quality Assurance Program

Safety Criterion: 7.3 – 1

Applicable Project Phases - All

A Quality Assurance Program, as defined in the Quality Assurance Manual (QAM) shall be developed, submitted for DOE approval, and implemented.

Implementing Codes and Standards

ASME NQA-1-1989, *Quality Assurance Program Requirements for Nuclear Facilities*
10 CFR 830, Subpart A, Quality Assurance Requirements

Regulatory Basis

10 CFR 830 Subpart A	*Quality Assurance Requirements*	
DOE/RL-96-0006	*4.1.1.6*	*Defense in Depth-Human Aspects*
DOE/RL-96-0006	*4.1.4.1*	*Safety/Quality Culture-Safety/Quality Culture*
DOE/RL-96-0006	*4.1.6.1*	*Quality Assurance-Quality Assurance Application*
DOE/RL-96-0006	*4.1.6.2*	*Quality Assurance-Established Techniques and Procedures*
DOE/RL-96-0006	*4.1.6.3*	*Quality Assurance-Established Techniques and Procedures*

7.4 Unreviewed Safety Questions

Safety Criterion: 7.4 - 1

Applicable Project Phases - Operations (including Hot Commissioning) and Deactivation

An Unreviewed Safety Question program shall be established and implemented in accordance with 10 CFR 830.203.

Implementing Codes and Standards

DOE G 424.1-1, "Implementing Guide for Use in Addressing Unreviewed Safety Question Requirements." (DOE G 424.1-1 as tailored in the WTP specific USQ procedure submitted pursuant to 10 CFR 830.203 (d) and approved by the ORP as part of the Authorization of Hot Commissioning regulatory action.)

Regulatory Basis

10 CFR 830 Subpart B	*Safety Basis Requirements*	
DOE/RL-96-0006	*4.4.4*	*Unresolved Safety Questions*

7.5 Conduct of Operations

Safety Criterion: 7.5 – 1

Applicable Project Phases – Cold Commissioning, Operations (including Hot Commissioning) and Deactivation

The conduct of operations program shall be established and implemented using a tailored approach and addressing the following:

1 Operations organization and administration. Facility polices will describe the philosophy of standards of excellence under which the facility is operated and clear lines of responsibility for normal and emergency conditions established. The direct responsibility for process safety rests with the contractor. The facility manager will ensure that all elements for safety facility operation are in place, including an adequate number of qualified and experienced workers. The minimum requirements will be set for the availability of staff and equipment. The operating organizations shall become and remain familiar with the features and limitations of components included in the design of the facility. They shall obtain appropriate input from the design organization on the planning and conduct of training.

2 Shift routines and operating practices. Standards for the professional conduct of operations personnel will be established and followed, so that operator performance meets expectations of facility management. On-shift operating crew will operate the facility through adherence to operating procedures and technical safety requirements and sound operating practices. Normal operation, including anticipated operational occurrences, maintenance, and testing, shall be controlled so that facility and system variables remain within their normal operating ranges and the frequency of demands placed on Important to Safety structures, systems, and components are small.

3 Control area activities

4 Communications

5 Control of on-shift training

6 Investigation of abnormal events

7 Notifications

8 Control of equipment and system status. The facility is required to establish administrative control programs to handle configuration changes resulting from maintenance, modifications, and testing activities. Not only must the operating shift be aware of how equipment, and systems will function for operational purposes, but in order to satisfy the design bases and the operational limits, the proper component, equipment, and system configuration must be established and maintained.

9 Lockout and tagout

10 Independent verification

11 Logkeeping

12 Operations turnover

13 Operations aspects of facility chemistry and unique processes. Operators should understand the responsibilities associated with their positions (both in process monitoring and control and in interface with the technical process department). Operators should be knowledgeable about aspects of facility processes and safety that affect operation and should be able to analyze off-normal

situations and take appropriate action to correct the cause(s) of the problem. They should obtain appropriate input from the design organization on pre-operational testing, operating procedures, and the planning and conduct of testing.

14 Required reading

15 Timely orders to operators

16 Operations procedures. Operating procedures will provide specific direction for operating systems and equipment during normal and postulated abnormal and emergency conditions. Operating procedures should provide appropriate direction to ensure that the facility is operating within its design bases and should be effectively used to support safe operation of the facility.

17 Operator aid postings

18 Equipment and piping labeling

Implementing Codes and Standards

DOE Order 5480.19, *Conduct of Operations Requirements for DOE Facilities*, as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006	*4.1.1.3*	*Defense in Depth-Control*
DOE/RL-96-0006	*4.1.2.2*	*Safety Responsibility-Safety Assignments*
DOE/RL-96-0006	*4.1.5.2*	*Configuration Management-Contractor Design Knowledge*
DOE/RL-96-0006	*4.3.1.1*	*Conduct of Operations-Organizational Structure*
DOE/RL-96-0006	*4.3.1.2*	*Conduct of Operations-Normal Operations*
DOE/RL-96-0006	*4.3.1.3*	*Conduct of Operations-Emergency Operating Procedures*
DOE/RL-96-0006	*4.3.1.4*	*Conduct of Operations-Readiness*
DOE/RL-96-0006	*5.1.3*	*Process Safety Responsibility*

7.6 Maintenance

Safety Criterion: 7.6 – 1

Applicable Project Phases - Cold Commissioning, Operations (including Hot Commissioning), and Deactivation

The maintenance program shall contain a DOE-approved Maintenance Implementation Plan (MIP) and be developed using a tailored approach addressing each of the following elements:

1 Organization and administration
2 Maintenance training and qualification
3 Maintenance facilities, equipment, and tools
4 ~~Types of maintenance.~~
5 Maintenance procedures and other work-related documents
6 Planning, scheduling, and coordinating maintenance activities
7 Control of maintenance activities
8 Post-maintenance testing
9 Procurement of parts, materials, and services
10 Material receipt, inspection, handling, storage, retrieving, and issuance
11 Control and calibration of measuring and test equipment
12 Maintenance tools and equipment control
13 Documented facility condition inspections to identify and address aging effects
14 Management involvement with facility operations
15 Maintenance history and trending
16 Analysis of maintenance-related problems
17 Modification work

Appropriate, regular preventive maintenance, inspection, and testing and servicing shall be performed to preserve, predict, and restore the availability, operability, and reliability of Important to Safety (ITS) SSCs. The program shall maintain ITS SSCs to assure that reliability targets for system and components to start or run are met, when such values are credited in the safety analysis. The program shall also assure that mechanical integrity of ITS process equipment and SSC's is maintained.

Implementing Codes and Standards

DOE Order 433.1, *Maintenance Management Program for DOE Nuclear Facilities*, as tailored in Appendix C

Regulatory Basis

DOE/RL-96-0006	*4.2.7.1*	*Reliability, Availability, Maintainability, and Inspectability (RAMI)-Reliability*
DOE/RL-96-0006	*4.3.5.1*	*Operational Testing, Inspection, and Maintenance*
DOE/RL-96-0006	*5.2.7*	*Mechanical Integrity*

7.7 Reporting and Incident Investigation

Safety Criterion: 7.7 – 1

Applicable Project Phases - All

An incident and reporting investigation program shall be developed, documented, and implemented that:

1 Investigates each incident which results in, or could reasonably have resulted in, a major accident

2 Conducts the investigation promptly

3 Develops, recommends and implements appropriate corrective measures

4 Submits results of the investigation to the DOE via ORPS database for evaluation and in support of regulatory oversight

Implementing Codes and Standards

DOE Manual M 231.1-2, Attachment 2, *Occurrence Reporting and Processing of Operations Information*

Regulatory Basis

DOE/RL-96-0006 *5.2.10 Incident Investigation*

Safety Criterion: 7.7 – 2

Applicable Project Phases - All

Facility management shall institute measures to ensure that events relevant to safety are detected and evaluated, and that necessary corrective measures are taken promptly and information from them is disseminated in accordance with the requirements of DOE M 231.1-2, Attachment 2. Operational event reports shall be prepared and submitted to the DOE. The facility management shall have access to operational safety experience from other related facilities.

Implementing Codes and Standards

DOE Manual M 231.1-2, Attachment 2, *Occurrence Reporting and Processing of Operations Information*

Regulatory Basis

DOE/RL-96-0006 *4.3.1.8 Conduct of Operations – Operational Events*

7.8 Emergency Preparedness

Safety Criterion: 7.8 - 1

Applicable Project Phases - Cold Commissioning, Operations (including Hot Commissioning) and Deactivation

An emergency management program shall be developed, documented, and implemented for the purpose of protecting public health and the environment. The emergency management program shall be documented in a facility-specific emergency plan which is an integral part of the Hanford Site Emergency Preparedness documentation hierarchy. The facility-specific plan, together with the *Hanford Site Emergency Management Plan* (DOE/RL-94-02), will address the following program elements:

(1) The establishment and maintenance of a facility emergency response organization with clearly specified authorities and responsibilities for emergency response and mitigation.

(2) Provisions for interfaces and coordination with Hanford Site and offsite agencies in the areas of planning, preparedness, response, and recovery.

(3) A description of the hazards and potential consequences resulting from analyzed accidents.

(4) Identify and describe the capabilities for detection of emergency events, the methodology for determining event severity, and the basis for declaring an emergency.

(5) The methods to be used to provide notification of an emergency event to Hanford Site organizations; offsite response agencies; and Federal, state, and local regulatory agencies.

(6) Provisions for assessing the consequences resulting from the release of hazardous materials.

(7) A description of protective actions for responders, workers, and the public, to include provisions for sheltering, evacuation, and personnel accountability.

(8) Medical support during emergency response, to include provisions for ambulance and hospital services and decontamination of injured personnel.

(9) Methodology for the safe shutdown of the facility, reentry to the facility during or after emergency response, and provisions for developing a recovery strategy following an accident.

(10) Public information program designed to provide the public, media, and employees with accurate and timely information.

(11) Adequate emergency facilities and equipment to support response.

(12) A training program will be designed to ensure that personnel are prepared to respond to, manage, mitigate, and recover from emergencies associated with facility operations.

(13) Emergency plans shall be prepared before the start of cold commissioning of the facility, and shall be exercised periodically to ensure that protection measures can be implemented in the event of an accident that results in, or has the potential for, unacceptable releases of radioactive materials within and beyond the facility control perimeter.

(14) Provisions for the administration of the program, to include a designated program administrator; program assessment and issue resolution; the development and maintenance of technical support documents, plans, and procedures; the coordination of activities; and maintenance of appropriate auditable records.

Implementing Codes and Standards

DOE/RL-94-02, *Hanford Emergency Management Plan*, as tailored in Appendix C

Regulatory Basis

WAC 173-303	*Dangerous Waste Regulations*	*Location: Part 350*
WAC 246-247	*Radiation Protection - Air Emissions*	*Location: Part 075 (12)*
DOE/RL-96-0006	*4.1.2.3 Safety Responsibility - Site and Technical Support*	
DOE/RL-96-0006	*4.3.3.1 Emergency Preparedness - Offsite Measures*	
DOE/RL-96-0006	*4.3.3.2 Emergency Preparedness - Accident Management Strategy*	
DOE/RL-96-0006	*4.3.3.3 Emergency Preparedness - Establishment and Continued Exercise of Emergency Plans*	
DOE/RL-96-0006	*5.2.11 Emergency Planning and Response*	

8.0 Deactivation and Decommissioning

Safety Criterion: 8.0 – 1

Applicable Project Phases - All

There shall be an approved plan for deactivation of the facility before it is constructed. The objectives of the plan shall be to reduce radiation exposure to Hanford Site personnel and the public both during and following deactivation and decommissioning activities and to minimize the quantity of radioactive waste generated during deactivation, decontamination, and decommissioning. Features and procedures that simplify and facilitate decommissioning shall be identified during the planning and design phase based upon a proposed decommissioning method.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix F, "Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning"

Regulatory Basis

DOE/RL-96-0006 *4.2.3.3* *Radiation Protection-Deactivation, Decontamination, and Decommissioning Design*
DOE/RL-96-0006 *4.3.2.3* *Radiation Protection-Final Deactivation Plans and Provisions*

Safety Criterion: 8.0 – 2

Applicable Project Phases - All

Facilities shall be designed to simplify decontamination and decommissioning, reduce exposure to site personnel and the public during these activities, and increase the potential for reuse. Features and procedures that simplify and facilitate decontamination, decommissioning, and minimization of contaminated equipment and the generation of radioactive waste during deactivation, decontamination, and decommissioning shall be identified during the planning and design phase based upon a proposed decommissioning method or conversion to other use.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix F, "Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning"
DOE G 441.1-2, *Occupational ALARA Program Guide*

Regulatory Basis

10 CFR 835 *Occupational Radiation Protection Location: 1002*
DOE/RL-96-0006 *4.2.3.3* *Radiation Protection-Deactivation, Decontamination, and Decommissioning Design*
DOE/RL-96-0006 *4.3.2.3* *Radiation Protection-Final Deactivation Plans and Provisions*

9.0 Documentation and Submittals

Safety Criterion: 9.0 – 1

Applicable Project Phases – Cold Commissioning

The results of the pre-startup safety review should be submitted to DOE for evaluation and in support of authorization decisions and regulatory oversight.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix J, "Ad Hoc Implementing Standard for Startup"

Regulatory Basis

DOE/RL-96-0006	5.2.6	Pre-startup Safety Review

Safety Criterion: 9.0 – 2

Applicable Project Phases - All

The Contractor should request authorization for construction only after being satisfied by appropriate internal assessments that the main safety issues have been satisfactorily resolved and that the remainder are amenable to solution before operations are scheduled to begin.

A Preliminary Safety Analysis Report (PSAR) shall be submitted to the regulator only after all major safety issues have been resolved and other safety issues scheduled for completion. The PSAR shall document the facility design and plans for construction and demonstrate adequate planning for the operational phase.

A Final Safety Analysis Report (FSAR) shall be submitted to the regulator for approval prior to the authorization to operate the facility. The FSAR shall document the completed design and construction and provide details on the plans for operation. The FSAR shall include facility and process drawings and fabrication and construction specifications important to the safety analysis of the facility.

The FSAR shall identify significant changes made in the facility design and plans for operation from what was presented in the PSAR.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001, *Safety Requirements Document Volume II*, Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"
24590-WTP-QAM-QA-01-001, *Quality Assurance Manual*
Policy Q-18.1, "Independent Assessment (Audit)"

Regulatory Basis

DOE/RL-96-0006	*4.4.3*	*Recommendation for Initiation of Construction*
10 CFR 830		*Nuclear Safety Management*

Safety Criterion: 9.0 – 3

Applicable Project Phases - All

Material that is part of the authorization basis shall be established, documented, and submitted to the DOE for evaluation and in support of decisions and regulatory oversight. The material shall be maintained current with respect to changes made to the facility design and administrative controls and in the light of significantly new safety information.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"
Appendix I, "Ad Hoc Implementing Standard for Project Integrated Safety Management Approach"

9.1 Safety Analysis Reports

Safety Criterion: 9.1 – 1

Applicable Project Phases - All

Safety analyses shall be performed using a tailored approach to develop and evaluate the adequacy of the authorization basis for the facility. Preliminary and Final Safety Analysis Reports shall be prepared to document the safety analyses.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"

Regulatory Basis

DOE/RL-96-0006	*4.1.3.1*	*Authorization Basis-Authorization Basis*
DOE/RL-96-0006	*4.2.1.3*	*Design-Safety Analysis*

Safety Criterion: 9.1 – 2

Applicable Project Phases - All

A SAR shall contain sections that address the following topics:

(1) Site Description
(2) Facility and Process Description
(3) Integrated Safety Analysis
(4) Nuclear Criticality Safety
(5) Technical Safety Requirements
(6) Radiation Safety
(7) Chemical Safety
(8) Fire Safety
(9) Human Factors
(10) Emergency Preparedness
(11) Management Organization
(12) Conduct of Operations
(13) Procedures

(14) Training and Qualification
(15) Deactivation and Decommissioning
(16) Incident Investigations
(17) Records Management
(18) Audits and Assessments
(19) Quality Assurance
(20) Initial Surveillance and In-Service Testing
(21) Maintenance

The SAR should also contain an Executive Summary that provides an overview of the facility safety basis and presents information sufficient to establish a top-level understanding of the facility, its' operation, and the results of the safety analysis.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"

Safety Criterion: 9.1 - 3

Applicable Project Phases - All

All work concerning the facility shall be carried out in accordance with the approved SAR.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix I, "Ad Hoc Implementing Standard for Project Integrated Safety Management Approach"

Regulatory Basis

DOE/RL-96-0006	4.4.3, *Recommendations for Initiation of Construction*
10 CFR 830 Subpart B	*Safety Basis Requirements*

Safety Criterion: 9.1 – 4

Applicable Project Phases – Operations (including Hot Commissioning) and Deactivation

The Final Safety Analysis Report (FSAR) shall be reviewed annually and updated as necessary to ensure that the information is current, remains applicable, and reflects all changes implemented up to six months prior to the filing of the updated FSAR. The regulatory approval of any Unreviewed Safety Questions, and the material submitted to the regulator in support of that approval, shall be considered an addendum to the FSAR until the information is incorporated into the FSAR as part of the next periodic update.

Facilities in operation for one year or more will address the results of the experience feedback program for the facility. Additionally, relevant experience from other facilities both within DOE and from the commercial nuclear industry should be considered.

Implementing Codes and Standards

DOE Guide DOE G421.1-2, *Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*

Section 4.1.3, "Annual DSA Updates (830.202)", as tailored in *Safety Requirements Document Volume II*, Appendix C

Regulatory Basis

DOE/RL-96-0006 *4.1.3.1 Authorization Basis-Authorization Basis*

9.2 Technical Safety Requirements

Safety Criterion: 9.2 – 1

Applicable Project Phases – Operation (including Hot Commissioning) and Deactivation

Technical safety requirements shall be prepared and submitted for approval, and the facility shall be operated in accordance with the approved technical safety requirements.

Implementing Codes and Standards

10 CFR Part 830, "Nuclear Safety Management"

Paragraph 830.205, "Technical Safety Requirements", items (a)(1) and (a)(2)

Subpart B, "Safety Basis Requirements", Appendix A, "General Statement of Safety Basis Policy", section G, items 1, 3, 4, and 5

Regulatory Basis

DOE/RL-96-0006 *4.1.3.1 Authorization Basis-Authorization Basis*

10 CFR 830, Subpart B *Safety Basis Requirements*

Safety Criterion: 9.2 – 2

Applicable Project Phases – Operation (including Hot Commissioning) and Deactivation

Technical safety requirements shall be based on the Final Safety Analysis Report and any additional safety requirements established for the facility.

Implementing Codes and Standards

10 CFR Part 830, "Nuclear Safety Management"

Paragraph 830.205, "Technical Safety Requirements", items (a)(1) and (a)(2)

Subpart B, "Safety Basis Requirements", Appendix A, "General Statement of Safety Basis Policy", section G, items 1, 3, 4, and 5

Regulatory Basis

10 CFR Part 830, Subpart B *Safety Basis Requirements*

Safety Criterion: 9.2 – 3

Applicable Project Phases – Operations (including Hot Commissioning)

Technical safety requirements shall consist of the following:

1 Safety limits
2 Operating limits
3 Limiting control settings
4 Limiting conditions for operation
5 Surveillance requirements
6 Administrative controls
7 Use and Application provisions
8 Design features
9 Bases Appendix

Implementing Codes and Standards

10 CFR Part 830, "Nuclear Safety Management"
Subpart B, "Safety Basis Requirements", Appendix A, "General Statement of Safety Basis Policy", section G, items 4, 6, and Table 4

Regulatory Basis

DOE/RL-96-0006 *4.3.1.6 Conduct of Operations-Operations Within the Authorization Basis*
10 CFR 830, Subpart B *Safety Basis Requirements*

Safety Criterion: 9.2 – 4

Applicable Project Phases – Operations (including Hot Commissioning) and Deactivation

Technical safety requirements shall be kept current at all times so that they reflect the facility as it exists and as it is analyzed in the Safety Analysis Report.

Implementing Codes and Standards

10 CFR Part 830, "Nuclear Safety Management"
Subpart B, "Safety Basis Requirements", Appendix A, "General Statement of Safety Basis Policy", section G, item 5

Regulatory Basis

DOE/RL-96-0006 *4.1.3.1 Authorization Basis-Authorization Basis*
10 CFR 830, Subpart B *Safety Basis Requirements*

Safety Criterion: 9.2 – 5

Applicable Project Phases – Operations (including Hot Commissioning) and Deactivation

All proposed revisions to technical safety requirements shall be submitted for regulatory approval prior to implementation of the revision. The submission shall include the basis for the proposed revision. Revisions to the bases sections can be made without DOE approval if the changes are editorial in nature and do not make significant changes.

Implementing Codes and Standards

10 CFR Part 830, "Nuclear Safety Management"
Paragraph 830.205, "Technical Safety Requirements", item (a)(2)
Subpart B, "Safety Basis Requirements", Appendix A, "General Statement of Safety Basis Policy", section G, items 5 and 6

Regulatory Basis

10 CFR Part 830, Subpart B, Safety Basis Requirements

Appendix A

Implementing Standard for Safety Standards and Requirements Identification

CONTENTS

1.0 Introduction

This standard implements the process for establishing a set of radiological, nuclear, and process safety requirements and standards as described in DOE/RL-96-0004 and RL/REG-98-17. The Project refers to this process as Integrated Safety Management (ISM).

The activities described below establish radiological, nuclear and process safety standards and requirements for design, construction, and operation of the facility. Establishment of safety standards and requirements is an iterative process that takes place throughout the life of the project. The process repeatedly evaluates these standards and requirements based on the evolving design.

The Safety Requirements Document (SRD) provides formal documentation of the standards resulting from this process. Structures, systems, and components (SSCs) that serve to provide reasonable assurance that the facility can be operated without undue risk are classified as Important to Safety and are defined in Safety Criterion 1.0-6. For specific SRD safety criteria implementing codes and standards are specified for safety design class, safety design significant, safety class, and safety significant SSCs. For specific SRD safety criteria implementing codes and standards for risk reduction class (RRC) and additional protection class (APC) SSCs shall be specified using the process set forth in this SRD Appendix A ISM process (i.e., the implementing standard for safety standards and requirements identification to meet DOE/RL-96-0004) and need not otherwise be specified in the SRD with one exception: For appendices to the SRD designated as "implementing standards", provisions of these appendices specified for RRC and APC SSCs remain in effect. The SRD is updated as needed to reflect the results of successive iterations of the standards and requirements identification process (i.e., the ISM process). This paragraph is applicable only to the following Safety Criteria: 4.1-1, 4.1-2, 4.1-3, 4.1-4, 4.2-1, 4.2-2, 4.2-3, 4.3-1, 4.3-2, 4.3-3, 4.3-4, 4.3-5, 4.3-6, 4.3-7, 4.4-1, 4.4-2, 4.4-3, 4.4-4, 4.4-5, and 4.4-6. However, for Safety Criteria 4.1-2 and 4.1-3, the implementing codes and standards contained in these safety criteria shall be applicable to APC SSCs designated SC-II or SC-III as they apply to seismic performance.

2.0 Process Initiation

The WTP Project Manager shall ensure implementation of the Project Management Plan, thus assuring that adequate resources are available and organized to perform the tasks required by this standard. Personnel with appropriate technical backgrounds shall be assigned to the tasks. This activity also assures that the input information required for the safety standards and requirements identification process has been collected and organized. This input information includes the top-level safety standards and principles stipulated by DOE in DOE/RL-96-0006 and the laws and regulations applicable to the WTP project.

The DOE/RL-96-0004 safety requirements and standards identification Process Manager for the project is the Radiological, Nuclear, and Process Safety Manager.

The Process Manager chairs the DOE/RL-96-0004 safety requirements and standards identification Process Management Team (PMT). The PMT is constituted in accordance with project implementing documents and includes managers from the following project organizations:

- Environmental and Nuclear Safety
- Engineering
- Operations

The Process Management Team shall oversee the ISM process and shall provide resources and resolve issues as necessary. The PMT shall set up ISM Teams for the conduct of ISM usually on a plant system basis. During facility operation, the process hazard analysis shall be updated to reflect changes concurrently with the annual update of the FSAR. In addition the process hazard analysis will be updated and submitted to the Office of River Protection as required by RL/REG-97-13, *Office of River Protection Position of Contractor-Initiated Changes to the Authorization Basis*. Individual PMT members shall provide various subject matter experts to help fulfill the roles required of the ISM Teams for conduct of the ISM process.

3.0 Identification of Work

The aim of this activity is to describe the work that will be performed so that the hazards inherent in the work can be identified and evaluated. Work activity experts who have extensive knowledge of the overall processing approach and are integrally associated with the facility design shall perform this activity. Work activity experts shall be drawn from the following WTP organizations:

- Engineering staff
- Operations staff

When appropriate, the PMT may also draw work activity experts from the staff of other departments, such as from Construction.

In an overall sense, identification of work involves definition of the project mission and identification of the processes that must be performed to accomplish the mission. It includes selection of optimum functions, processes, and parameters through trade studies and definition of functional requirements. Identification of work for the purpose of design development involves definition of various plant systems, structures, and components. This latter definition is the focus for the ISM Teams created to conduct ISM on a plant system basis.

The product of this activity includes:

- Process description
- System descriptions
- Descriptions of key structures
- Basis of design documents
- PFDs, MFDs, and P&IDs

4.0 Hazard Evaluation

The aim of the hazard evaluation activity is to identify and characterize the hazards resulting from the work. The ISM Teams shall conduct the hazard evaluation activity on a plant system basis. These teams shall include work activity experts (as defined in section 3.0), hazard assessment experts, and hazard control experts.

Hazard assessment experts and hazard control experts shall generally be members of the technical staffs of the Safety Analysis Manager and of the Regulatory Safety Manager. The process management team shall provide additional technical resources as required to evaluate the hazards.

The hazard evaluation shall address hazards inherent in normal operation as well as potential accidents resulting from abnormal internal and external events.

The hazard evaluation shall comprise the following elements:

- Identification of Hazards
- Identification of Potential Accident/Event Sequences
- Estimation of Consequences
- Estimation of Event Frequencies
- Consideration of Dependent Failures
- Selection and Analysis of Design Basis Events
- Definition of Operating Environment
- Identification of Potential Control Strategies
- Documentation of the Hazard Evaluation

These elements are discussed below.

4.1 Identification of Hazards

The objective of this element is to systematically identify the hazards associated with the defined work.

The ISM Teams shall compile a list of hazardous materials and energy sources associated with the facility processes, design, and operations. This list shall be compiled based on the identified work. This compilation provides information used to identify potential accidents resulting in the uncontrolled release of hazardous material or energy to facility or co-located workers, the public, and the environment. The team may use checklists to guide the compilation process and to assure that all potential hazards from both natural and manmade sources originating from outside and inside the facility are addressed.

4.2 Identification of Potential Accident/Event Sequences

The objective of this element is to perform a structured and systematic examination of the facility and its operations to identify potential accidents (including those resulting from common mode and common cause failures). The team shall conduct this examination using methodologies and guidelines in AIChE (1992).

4.3 Estimation of Consequences

4.3.1 Accident Severity Level Identification

A severity level, SL, shall be assigned to each postulated radiological accident with co-located worker and public receptor consequences. The severity level shall reflect the unmitigated consequences of the postulated accident (i.e., should not credit SSCs that prevent or mitigate the release) with the following exception. The severity level assignment may credit the contribution that a cell or cave makes to a leak path factor, to limitation of spilled liquid pool size, or to plateout when the credited aspect of the cell or cave is not challenged by the event. Consequence estimates supporting severity level assignment shall be based on bounding assumptions regarding such factors as quantity, form, leak path, plateout, and location of the radioactive material available for release, and the energy sources available to interact with the hazardous material. Severity level consequence estimates shall be evaluated as ground level releases. The severity level shall be defined as follows:

SL	Facility Worker Consequence*	Collocated Worker Consequence	Public Consequence
SL-1	> 100 rem/event	> 100 rem/event	> 5 rem/event
SL-2	5 - 100 rem/event	5 - 100 rem/event	1 - 5 rem/event
SL-3	1 - 5 rem/event	1 - 5 rem/event	0.1 - 1 rem/event
SL-4	< 1 rem/event	< 1 rem/event	< 0.1 rem/event
*The column for "Facility Worker Consequences" does not apply to SC, SS, or APC SSCs.			

Facility Worker Consequence Determination*

Facility Worker Consequence Ranking	Qualitative Criteria
High	Prompt worker fatality or serious injuries (e.g., immediately life threatening or permanently disabling) or significant radiological or chemical exposures. • >100 rem • >ERPG-3

Facility Worker Consequence Determination*

Moderate	Injuries that might require hospitalization but are not immediately life-threatening and are not permanently disabling • 25-100 rem • ERPG-2 to ERPG-3
Low	Less than moderate consequences
Standard Industrial	1. Other hazards typically encountered in the nuclear and chemical industry, regardless of potential consequences. 2. Other hazards for which national or regional regulatory bodies exist outside of the DOE (e.g., OSHA)

* This table does not apply to SDC, SDS, or RRC SSCs.

Consequences to the co-located worker and the public shall be evaluated at the locations specified in the *Safety Requirements Document, Volume II, Safety Criterion 2.0-1.*

Early in the design, the severity level estimate may be quantitative analysis or a qualitative assessment based on the experience of the ISM Teams. Assumptions upon which the severity level estimates are based shall be documented and linked by reference to the hazardous situation to which they apply. As the design progresses, formal accident analyses are performed as described in Section 4.3.2. These accident analyses do not address all of the potential accidents identified, but they do address bounding events. As the design progresses, early assumptions may be confirmed or replaced by design information. If later design information changes the conclusion of the severity level assessment, the effect of the change on subsequent activities of the ISM process shall be evaluated by the ISM Team.

Severity level designations are not required for postulated accidents that have only facility worker consequences for SC, SS, or APC SSCs. For these situations, facility worker consequences are estimated based on qualitative evaluation at the worst-case occupied location.

The potential consequences of releases of hazardous chemicals shall also be assessed. The assessment shall consider both the inherent hazard of the chemical itself, and the potential for the chemical hazard to initiate or exacerbate a radiological hazard.

4.3.2 Accident Analysis

Accident analyses provide confirmation of the estimates of accident consequences made by the ISM Teams (Section 4.3.1) and confirm the selection of the preferred hazard control strategies (Section 5.0).

The formal accident analyses shall address internal design basis events, man-made external events, and natural phenomena hazards.

The postulated internal events shall be grouped by type. Potential groupings include the following:

- Liquid spills

- Spills of solid materials
- Pressurized releases
- Chemical reactions
- Boiling
- Flammable gas ignition
- Fires
- Load drops
- Criticality (prevented)

The accident analysis shall consider the following factors to the extent they are important to the scenario in consideration:

- The quantity and nature of the material at risk.
- The respirable release fraction.
- The fraction of the airborne material released to potentially occupied locations or the environment.
- Atmospheric dispersion.
- Radiological composition of the material released.
- External radiation field.
- Exposure times.

The accident analysis shall address the potential consequence to facility workers, co-located workers, and the public. For facility workers, quantitative assessment of consequences as part of the hazards analysis (ISM) process is appropriate and sufficient. In unique instances, quantitative calculation of worker consequences may be required to further define a hazardous situation in support of the ISM control selection process.

4.3.3 Normal Conditions

Some hazards inherent in normal operation must be mitigated to comply with the standards for normal operation in SRD Chapter 2.0. Such hazards shall be addressed in accordance with the WTP Radiation Protection Plan.

4.4 Estimation of Event Frequencies

There is normally insufficient information early in the design to accurately quantify the frequency of postulated internal events because this frequency depends on the design of the SSCs that implement the control strategy used to manage the hazard. At an early stage, frequency evaluations may be based on the team's experience with similar hazards in similar facilities. The team shall validate these estimates as the design develops.

As the design matures, information on the frequency of hazardous events may be gained from the use of hazard evaluation techniques that provide frequency data (e.g., event and fault trees). Evaluations of the frequency of failure in redundant systems or in diverse systems using similar equipment shall consider dependent failures.

The frequencies of design basis external events may be derived from existing analyses (e.g., safety analyses for adjacent facilities), from evaluation of historical data (e.g., transportation data), or from site-specific information (e.g., seismic history).

4.5 Consideration for Dependent Failures

The potential for dependent failure mechanisms shall be identified and considered during the estimation of accident frequencies when seeking control strategies. Without such consideration, the results may be potentially non-conservative (i.e., result in unjustifiably optimistic predictions of accident frequencies or process reliabilities, given the selected strategies).

Three broad categories of dependencies are used to classify and define the dependent failures that are expected to be important to the WTP project. Each represents a functionally different way in which commonalties between redundant systems, trains, or components can potentially reduce their overall expected reliability and are defined as follows:

- Functional dependencies
- Spatial dependencies
- Institutional dependencies

Functional Dependencies. These dependencies reflect the reliance of multiple systems, trains, or components on a single system, train, component, or process condition. These dependencies typically result from:

- Process upsets that present simultaneous challenges to redundant systems, trains, or components.
- Failure of individual components that provide multiple functions.
- Failure of individuals components that are shared by otherwise independent trains or systems.
- Failure of common support systems that provide motive power, cooling, control, and actuation of process and safety components throughout the facility.
- Dependent system failures which result from operator error, where the operator is serving as a system control element.

Spatial Dependencies. Spatial dependencies between otherwise independent pieces of equipment originate with their relative locations and the potential for physical interactions or common loss.

Examples include the near simultaneous failure of two components as a result of their co-location in an area that experiences the effects of:

- Internal fires or explosions.
- Internal floods from such equipment as failed tanks and cooling systems.

- Externally applied forces and loads from such events as seismic activity, airplane crashes and vehicle crashes.
- Natural forces and environmental conditions, e.g., severe weather, lightning, floods, and external fires.

Institutional Dependencies. Institutional dependencies come from activities within the plant which are conducted by maintenance workers, operators, designers, and equipment procurers that result in the near-simultaneous failure of otherwise independent components. These may also be called common cause failures because their effect is often manifest as a set of components failing in the same way at approximately the same time. Examples of the causes for failure of this type include:

- Use of identical components with the same maintenance and operating cycle that contributes to near simultaneous wear-out.
- Use of identical components that lead to the appearance of coincident failures resulting from inherent design weaknesses or from the misapplication of hardware (improper service factor).
- Labeling, training, procedural, and administrative control inadequacies that allow, or cause, operators/maintenance workers to make the same or similar errors on more than one system, train, or component.
- Using a single maintenance crew to maintain/adjust/calibrate independent equipment during the same time period (a mistake/error during the maintenance or restoration of one piece of equipment is repeated on a second, similar piece of equipment so that the probability of near simultaneous failure is increased).

4.6 Selection and Analysis of Design Basis Events

The hazard evaluation performed by the ISM Team involves the identification of internal hazards and hazardous situations leading to the selection of a set of internal design basis events. These design basis events shall be selected to establish a set of bounding performance requirements for the SSCs relied upon to control the internal hazards and hazardous situations. Analysis of the design basis events also provides confirmation that the design satisfies the requirements of SRD Volume II Safety Criteria 2.0-1 and 2.0-2.

The hazard evaluation shall also select a set of external man made design basis events based upon information provided to the ISM Team on nearby facilities and transportation. These events shall establish a set of bounding performance requirements for the SSCs relied upon to mitigate these external events.

Design basis natural phenomena loads shall be as defined in the SRD Volume II Safety Criterion 4.1-3.

4.7 Definition of Operating Environment

The hazard evaluation shall define a set of bounding operating conditions in which SSCs relied upon to control hazards must function. Environmental parameters to be addressed include the following:

- Temperature

- Pressure
- Humidity
- Radiation Levels
- Chemical Environment

4.8 Identification of Potential Hazard Control Strategies

Based on the experience and judgement of team members, the ISM Team shall identify one or more potential hazard control strategies to manage each potential accident (i.e., hazardous situations that may result in unacceptable consequences). This set of potential hazard control strategies shall address means of preventing the potential accident and should address means of mitigating the consequences of the accident. The function(s) of each potential hazard control strategy should be clearly described. Potential hazard control strategies shall be identified to manage accident conditions arising from upsets in the process, conditions arising from external events, and conditions inherent in the normal operation of the process.

4.9 Documentation of the Hazard Evaluation

The results of the hazard evaluation shall be documented in the safety analysis report (SAR). The results of the process of conducting the various steps of the hazard evaluation shall be contained or referenced in a hazard database. For each hazard considered, the hazard database shall record or reference the following information produced by the hazard evaluation:

- Hazard identifier
- Hazard description
- Initiators of the hazardous situation
- Hazard severity level estimates for the public and co-located workers. For SDC, SDS and RRC SSCs for facility workers, severity level estimates may be determined qualitatively. For SS and APC SSCs for facility workers, severity level estimates need not be done.
- Qualitative hazard consequence determination result for the facility worker
- Basis for the severity level assignment or qualitative hazard consequence determination result, including assumptions affecting the estimate
- Hazard frequency estimate
- Basis for frequency estimate
- Potential hazard control strategies and functional requirements
- References for the hazard (these would typically be products of the work identification process)

The SAR shall also contain information on the performance of the hazard evaluation. This information shall include the following:

- Description of the comprehensive approach to hazard evaluation
- Description of the methodology for identification and quantification of work hazards
- Description of the methodology for identifying potential accident scenarios

- Description of the methodology for consequence assessment
- Clear identification of assumptions (e.g., quantity and form of material at risk, rate of release and relevant process conditions) that may drive or inhibit the potential accident
- Evidence of appropriate staffing, and adequate technical staffing and structure applied to the hazard evaluation

5.0 Development of Preferred Hazard Control Strategies

The aim of this activity is to identify a means of controlling each of the hazards identified in the hazard evaluation. The ISM Teams that include work activity experts, hazard assessment experts, and hazard control experts, as discussed in Sections 3.0 and 4.0, perform this activity.

The PMT members shall provide additional technical resources as required to develop the preferred hazard control strategies.

The ISM Teams select preferred control strategies based on the set of potential controls identified by the hazard evaluation team. Selection of the preferred strategy considers the following factors:

- The functions required of the preferred hazard control strategy in order to control the hazard
- The degree of defense in depth and reliability provided by the preferred hazard control strategy. The Implementing Standard for Defense in Depth provides requirements and goals in this area.
- Applicable design basis events.
- The operating environment (e.g., temperature and humidity) in which the SSCs implementing the preferred hazard control strategy must function.
- Effectiveness and efficiency of the preferred hazard control strategy.
- Conformance with the DOE stipulated top level standards.
- Compliance with applicable laws and regulations.

The preferred hazard control strategy should be documented in the SAR and will typically comprise a series of elements including some or all of the following:

- Passive and/or active SSCs that function to prevent the release (that is, SSCs that reduce the probability that a release will occur)
- Passive and/or active SSCs that function to mitigate the release (that is, SSCs that reduce the consequences once a release has occurred)
- Administrative controls (for example, limits on inventory)

Consistent with the defense in depth principle, the control strategy development should emphasize preventive measures. It should also emphasize passive SSCs over active SSCs and retention of released material over dispersion. Ideally, the preferred control strategy should incorporate SSCs that prevent releases and SSCs that mitigate the consequences of a release, should it occur.

Once the preferred control strategy is identified, it shall be evaluated for the most bounding conditions (i.e., the most demanding requirements imposed by the set of hazardous situations that credit the function of the control strategy) using the techniques described in Section 4.3 through 4.5. In addition, the evaluation of the preferred hazard control strategy shall identify the measures necessary to assure that it performs its functions reliably. Such measures include maintenance requirements, testing intervals and calibration frequency. The results of this evaluation serve to confirm that the preferred hazard control strategy is capable of satisfying SRD Safety Criterion 2.0-1.

If credit is taken for operator action to satisfy the public radiological exposure standards of Safety Criterion 2.0-1, adequate radiation protection is provided to permit access and occupancy of the control room or other control locations under accident conditions without personnel receiving radiation doses in excess of 5 rem total effective dose equivalent (TEDE), 30 rem thyroid, and 30 rem beta skin for the duration of the accident. In the event operator action is not required, other than immediate actions required to place the facility operation into a safe state, then the worker exposure standards of Safety Criterion 2.0-1 apply. If credit is taken for operator action to satisfy public chemical exposure to the standards of Safety Criterion 2.0-2, provisions for operational access and control are made so that the operator exposure does not exceed the limits specified in Safety Criterion 4.3-7.

Documentation of the hazard control strategy development process shall clearly indicate selection of the preferred hazard control strategies and show the linkage of the control strategies to the respective hazards. The preferred control strategy should be described in terms of the safety functions required (e.g., limit release of radionuclides, etc.) and in terms of a set of engineered features, administrative controls (procedures and training), and management systems selected for implementing the strategy. When the nature of the hazard or hazardous situation is such that the appropriate preferred hazard control strategy is self-evident, the documentation need only demonstrate that the control strategy meets most, if not all, of the selection criteria, and need not provide a discussion of other, nonapplicable control strategies. Similarly, where a proven preferred hazard control strategy that is appropriate to the hazard exists and it is obvious to the team that there are no other alternative control strategies that could be equally attractive, then the documentation need only demonstrate that the control strategy meets most, if not all, of the selection criteria. Otherwise, the documentation should identify all control strategies considered and provide a defensible rationale for selection of the preferred strategy.

The following information produced by the preferred hazard control strategy definition shall be recorded in the hazard database:

- Preferred hazard control strategy
- Linkage of the preferred hazard control strategy to the respective hazards
- Rationale for preferred hazard control strategy selection
- Defense in depth provided
- Control strategy functions and performance requirements
- Estimate of the unmitigated event frequency
- Estimate of the consequences from the mitigated event (by performance of the Design Basis Event [DBE] analysis)
- Estimate of the mitigated event frequency (by performance of the DBE analysis)

- Applicable design basis events (e.g., design basis earthquake)

One of the issues in developing a preferred hazard control strategy for a particular hazard or hazardous situation is determining the number of layers of prevention and mitigation appropriate for the hazard. The preferred hazard control strategies shall conform to the requirements defined in the Implementing Standard for Defense in Depth. In addition, the following guidance shall be considered in developing preferred hazard control strategies.

5.1 Approach for Radiological Release Events

The general WTP design approach is to provide two confinement barriers against the release of radiological materials. For process systems, during normal operation the process vessels, piping and dedicated process vessel ventilation systems form the primary confinement barrier; the process cells and associated ventilation system form the secondary confinement barrier. Releases from the primary confinement are mitigated by the secondary confinement.

The mitigated or prevented consequences resulting from implementation of the control strategy must conform to SRD Safety Criterion 2.0-1.

5.2 Approach for Direct Radiation Exposure Events

The general WTP design approach is to provide one passive physical barrier against exposure to direct radiation. For radiological materials that are contained with the process cells, the cell shield wall usually provides this barrier. For radiological material inventories located out of cells, container shielding usually serves as this barrier.

The accident severity levels defined in section 4.3.1 for radiological release events also apply to radiation exposure events.

As was the case for radiological release events discussed in section 5.1, administrative controls alone may be credited as the controls that protect facility workers, when appropriate. Timely evacuation from the vicinity of the hazard is considered to be an administrative control.

5.3 Approach for Chemical Events

The potential consequences of hazardous chemicals shall also be assessed. The assessment shall consider both the inherent hazard of the chemical itself, and the potential for the chemical hazard to initiate or exacerbate a radiological hazard.

As many of the chemical hazards of the WTP are not unique to the facility, the selection of preferred hazard control strategies begins with the identification of what has been required and accepted as prevention and mitigation features for industrial plants with a similar chemical hazard. To implement this activity the ISM Team documents the types of prevention and mitigation features typically used at facilities with similar chemical hazards and comments on the appropriateness of the features for the WTP. Those that are appropriate for the WTP are identified as preferred hazard control strategies for preventing or mitigating the associated hazardous situation for the WTP.

If the chemical hazard for the WTP poses a chemical risk that is unique to the WTP, additional (or augmented) accident prevention and/or mitigation features shall be considered. Some unique aspects of the WTP that would drive this consideration are:

1 The chemical hazard does not exist in many other facilities such that the database of prevention and mitigation features is limited.
2 The method of physically containing the hazardous chemical at the vitrification plant is different from normal industry practice.
3 The facility worker at the vitrification plant might work closer to the hazard.
4 The vitrification plant facility workers have less opportunity to isolate themselves from the chemical release (e.g., in industry practice the chemical is usually stored outside but for the WTP it is stored inside a building with a difficult egress).
5 The chemical hazard may lead to a hazardous situation that could adversely impact the ability of the operators to maintain the facility in a safe state.

6.0 Classification of Structures, Systems, and Components

Structures, systems, and components that serve as preferred hazard control strategies are classified as Important to Safety and further classified into subcategories of Important to Safety in accordance with SRD Safety Criterion 1.0-6. The quality levels assigned to SSCs and the attributes of these quality levels are provided in the Quality Assurance Manual (BNI 2001).

Safety Structures, systems, and components means both safety structures, systems, and components and safety significant structures, systems, and components.

Safety-class structures, systems, and components (SC SSCs) means the structures, systems, or components, including portions of process systems, whose preventive or mitigative function is necessary to limit radioactive hazardous material exposure to the public, as determined from safety analyses.

For the WTP project, safety-class SSCs include:

- SSCs determined by safety analysis to perform a preventative or mitigative function necessary to limit the radiological release resulting in an SL-1 consequence to the public or limit the radiological consequences from and SL-1 event to the public[1].
- Support SSCs to safety-class SSCs if their failures can prevent a safety-class SSC from performing its safety functions[2]

[1] SL-1 events to the public are unmitigated events with public consequences greater than 5 rem. Consequences in this range meet the Evaluation Guidelines described in DOE G 420.1-1 and DOE STD-3009-94 (i.e., they are "in the rem range" for design or "challenge" or "approach" the 25 rem Evaluation Guideline).
[2] Support SSCs are those SSCs that are relied upon by the safety SSC to perform its intended safety function (e.g., electrical power sources for ventilation).

- SSCs determined by the criticality safety analysis to present an inadvertent criticality.

Safety-significant structures, systems, and components (SS SSCs) means the structures, systems, and components which are not desiginated as safety-class structures, systems, and components, but whose preventive or mitigative function is a major contributor of defense in depth and/or worker safety as determined from safety analyses.

For the WTP project, safety-significant SSCs include:

- SSCs determined by safety analysis to perform a preventive or mitigative function necessary to limit the radiological release resulting in an SL-1 event to the co-located worker or SL-2 event to the public or limit the radiological consequences from an SL-1 event to the co-located worker or SL-2 event to the public.
- SSCs determined by safety analysis to perform a preventive or mitigative function necessary to limit the chemical consequences from an event that exceed worker or public exposure standards in Safety Criterion 2.0-2.
- Support SSCs to safety-significant SSCs that prevent or mitigate accidents with the potential for significant onsite consequences should be classified as safety significant if their failures prevent a safety-significant SSC from performing its safety-function.[2]
- SSCs, determined by safety analysis, whose failure is estimated to result in a prompt worker fatality or serious injuries (e.g., loss of eye, loss of limb) or significant radiological exposures to workers.[3] SSCs for protection from standard industrial hazards are not safety-significant. Support SSCs to safety-significant SSCs that prevent or mitigate accidents with the potential for significant localized consequences need not be classified as safety-significant.
- SSCs determined by the safety analysis that are major contributors to defense in depth for protection of the public or co-located workers.
- SSCs determined by safety analysis to prevent or mitigate a facility worker hazard categorized as high.

Additional-protection class structures, systems, and components (APC SSCs) means important to safety SSCs that are neither safety-class nor safety-significant.

For the WTP project, APC SSCs include SSCs not designated as safety-class or safety-significant such as those that:

- Ensure the integrity of boundaries retaining significant amounts of radioactive materials.
- Ensure the integrity of boundaries retaining significant amounts of extremely hazardous chemicals.
- Contribute significantly to achieving the risk goals of Safety Criteria 1.0-2 and 1.0-3.

[3] This is neither an evaluation guideline nor a quantitative criterion. It represents a threshold of concern for which safety-significant SSC designation may be warranted. Estimates of worker consequences for the purpose of safety-significant SSC designation are not intended to require detailed analytical modeling, due to the uncertainties in analyses, especially for facility workers. Considerations should be based on engineering judgement of possible effects and the potential added value of safety-significant SSC designation.

- Bring the facility to a safe state. These SSCs may provide automatic system response to such events or may be SSCs such as monitors or alarms that alert operators to the necessity of taking manual action.
- Whose failure under NPH loads could prevent a SC, SS, or APC SSC with NPH safety function from performing that function.
- Whose failure under NPH loads by itself or in combination with one or more SSCs may result in loss of function of any emergency handling, hazard recovery, fire suppression, emergency preparedness, communication, or power system that may be needed to preserve the health and safety of workers and visitors.

In addition, APC SSCs include:

- SSCs determined by safety analysis to prevent or mitigate a facility worker hazard categorized as moderate.

7.0 Identification of Standards

Identification of standards is an iterative activity. Initially, the set of standards and requirements is derived from a general understanding of the hazards and hazardous situations inherent in the work. As the design evolves, the hazard evaluation and the development of the preferred hazard control strategies justify tailoring the set of standards to better fit the hazards.

The identification of engineering/design, manufacture/fabrication, and construction standards is performed by an ISM Team including work activity experts, hazard assessment experts, hazard control experts, as discussed in Sections 3.0 and 4.0, and standards experts. This ISM Team need not be the same team that performed the previous work identification and hazard evaluation activities. Identification of other standards (e.g., quality assurance, conduct of operations, etc.) will be performed by specially constituted teams formed by the PMT. The aim of this activity is to identify a tailored set of standards and requirements that will assure adequate safety when implemented.

The process management team shall provide additional technical resources as required to identify the standards.

Standards experts shall be drawn from the following WTP organizations:

- Staff of the Engineering Manager
- Technical staff of the Area Managers
- Technical staff of the E&NS Manager

The standards identified are evaluated and tailored for each control strategy based on compliance with applicable laws and regulations and conformance with the DOE-stipulated top level standards, plus the output of the preceding hazard evaluation and control strategy development steps. Typical considerations include the following:

- The severity level of the hazard
- The number of independent SSCs that comprise the preferred hazard control strategy
- The preferred hazard control strategy functions - recognizing that a specific control strategy may have multiple functions and serve to control multiple hazards
- The service (operating) environment (such as temperature and humidity)
- The applicable design basis events analysis
- The reliability required of the preferred hazard control strategy

Documentation of the standards and requirements identification process provides justification of the set selected and links each preferred hazard control strategy to its associated set of standards. The information generated during standards selection is retained in one or more databases for each preferred hazard control strategy:

- Preferred hazard control strategy
- Service environment
- Applicable design basis events
- Applicable standards
- Performance requirements
- Testing/calibration requirements
- In-service inspection requirements
- Maintenance requirements
- Quality level
- Standards justification

This information is structured so it can be linked to the preferred hazard control strategies in the hazard evaluation records. This provides a link from the hazards and hazardous situations through the preferred hazard control strategies to the standards. Not all of this information will be available early in the design. For example, it will not be possible to define maintenance and testing requirements until the design is mature.

As the standards are tailored, discrepancies with the current version of the SRD may arise. Such discrepancies shall be recorded. Formal changes to the SRD require approval from DOE.

8.0 Confirmation of Standards

Based on the recommendation of the PMT, the WTP Project Safety Committee (PSC) Chair requests the PSC to confirm the selected set of standards. The PSC defines a review approach, carries out the review, and documents the findings of the review. Resolution of PSC comments shall be documented.

9.0 Formal Documentation

Following confirmation by the PSC, the results of the standards selection process shall be documented in the SRD. The SRD shall incorporate documentation supporting these results by reference. The SRD shall identify and justify the set of requirements and standards selected to provide adequate protection of workers, the public, and the environment.

10.0 Recommendation

The recommended set of standards shall be certified in accordance with project implementing documents. When properly implemented, the set of standards:

1) Provides adequate safety
2) Complies with applicable laws and regulations, and
3) Conforms with the Top-Level Safety Standards and Principles

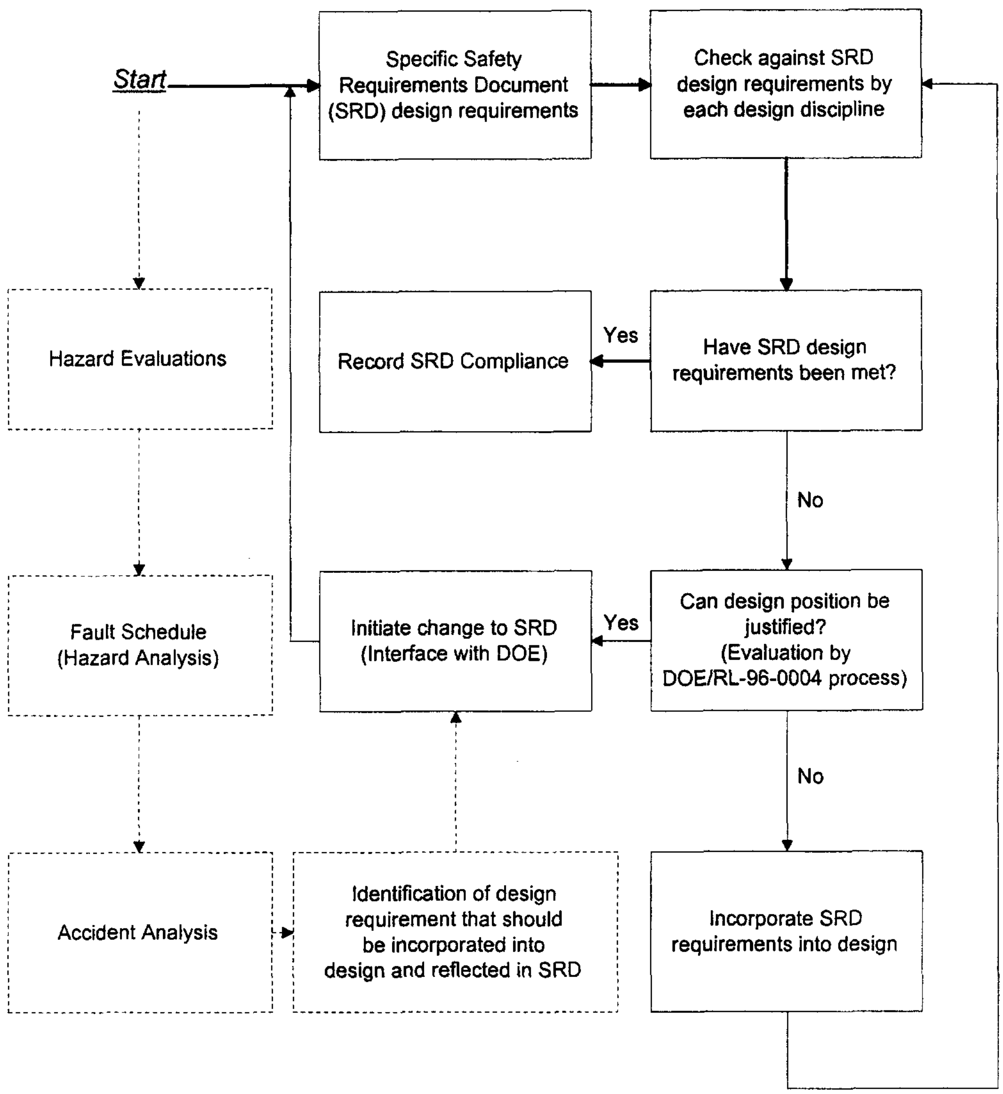
11.0 Maintenance of the SRD

Consistency of the SRD with current design information, hazards assessment, hazards control, and selected standards during the SRD development is ensured by participating with the personnel responsible for design and hazards analysis activities in the SRD development process as well as through reviews of the SRD, PSAR, and design information. Additionally, for design-related criteria, a review of the Safety Criteria against facility design will be conducted to ensure the Safety Criteria are met by the design. Figure A-1 depicts this process.

Proposed changes to the SRD are evaluated for impact on safety and compliance with regulations and the authorization basis (including hazard and accident analysis). These changes are then reviewed and approved commensurate with the process applied to the original configuration, including regulatory approval prior to implementing changes that could be considered as decreasing the level of safety. The essential elements of DOE/RL-96-0004, *Process for Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for the RPP Waste Treatment Plant Contractor*, as addressed in the original development of the SRD, are maintained, including the use of subject matter experts and the use of an equivalent level of review and approval of the proposed change.

After issuance of the construction approval, but prior to issuance of the SRD as part of the Operating Authorization Request package, the SRD will be controlled through the configuration management process. Additionally, DOE will be notified when the hazard analysis identifies a new situation affecting public safety or a significant revision occurs in a law or regulation that affects the design.

Figure A-1 SRD Compliance Process



12.0 Process for Updating Daughter Codes and Standards of Implementing Codes and Standards in the Safety Requirements Document (SRD)

This process for updating is applicable only to updating the national and industry consensus codes and standards used as daughter standards of implementing codes and standards in the SRD. It is not applicable to any changes to ad hoc standards, DOE directives, or other types of standards, or to changes to the parent implementing codes and standards.

1. An engineering evaluation will be performed on the updated standard. A reviewer knowledgeable of the standards in question shall:

   a. Ensure that the two versions of the standards are directly comparable and are substantially alike in philosophy and approach.
   b. Determine if any differences between the standards would result in changes that could preclude the applicable SSCs from performing their safety functions, or change the basis for the selection of the standard (or its parent standards).
   c. Document this determination, and its basis.

   An authorization basis change is not required if the differences do not result in changes that could preclude the applicable SSCs from performing their safety functions, or do not change the basis for the selection of the standard (or its parent standards).

   An authorization basis change is required using the standards selection process described in Appendix A of the SRD if the differences result in changes that could preclude the applicable SSCs from performing their safety functions, or change the basis for the selection of the standard (or its parent standards).

2. Engineering evaluations on updated standards will be approved by the Discipline Engineering Manager prior to the use of the updated standard for any quality affecting activity.

3. A listing of BNI-approved standard versions will be maintained as a controlled project quality document and a current copy provided monthly to DOE for information.

13.0 Definitions

Credible event: Any event with a frequency greater than 10^{-6} per year, including allowance for uncertainties.

Dependent Failures (Modarres 1993): In general, dependent failures are defined as events in which the probability of each failure is dependent upon the occurrence of other failures.

Important to Safety: Structures, systems, and components that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological, nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation).

This definition includes not only those structures, systems, and components that perform safety functions and traditionally have been classified as safety class, safety-related, or safety-grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems, or components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear, and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of important to safety, i.e., safety-related, may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems, and components should be considered for inclusion within this definition. The WTP has two classification schemes for Important to Safety SSCs. Both of the classification schemes are divided into three separate categories and are defined in Safety Criterion 1.0-6.

Mitigated event: As used in this standard, a mitigated event involves the following sequence:

- An initiating event that could lead to a release from the primary confinement barrier
- Failure of all elements of the control strategy that would prevent the initiating event from developing into a release from the primary confinement barrier
- Mitigation of the consequences of the release as provided by the control strategy

Mitigated event frequency: The mitigated event frequency is the product of the corresponding release frequency and the probability that the elements of the control strategy that mitigate the release will function given the release.

Release frequency: The release frequency is the product of the frequency of the initiating event and the probability that all elements of the control strategy that would prevent the release fail, given the initiating event.

Reliability: The probability that an SSC will perform its safety function when required.

Safe State: A situation in which the facility process has been rendered safe and no pressurized material flow occurs in the process lines. Any active, energy generating, process reactions are in controlled or passive equipment. The structures, systems, and components necessary to reach and maintain this condition are functioning in a stable manner, with all process parameters within normal safe state ranges.

Unmitigated event: As used in this standard, an unmitigated event involves the following sequence:

- An initiating event that could lead to a release from the primary confinement barrier
- Failure of all elements of the control strategy that would prevent the initiating event from developing into a release from the primary confinement barrier
- Failure of all elements of the control strategy that would mitigate the consequences of the release

Unmitigated event frequency: The frequency of an unmitigated event is the corresponding release frequency times the probability that all elements of the control strategy that would mitigate the release fail, given the release.

14.0 References

AIChE, 1992, American Institute of Chemical Engineers (AIChE), *Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples*, Center for Chemical Process Safety, New York, New York, USA, 1992.

BNI, 2001, *Quality Assurance Manual*, 24590-WTP-QAM-QA-01-001, Bechtel National, Inc., Richland, Washington, USA.

Modarres, 1993, *What Every Engineer Should Know about Reliability and Risk Analysis*, M Modarres, Marcel Dekker Inc., 1993, ISBN 0-8247-8958-X.

DOE/RL-96-0004, Revision 2, *Process of Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for the RPP Waste Treatment Plant Contractor.*

DOE/RL-96-0006, Revision 2, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for the RPP Waste Treatment Plant Contractor.*

Appendix B

Implementing Standard for Defense in Depth

CONTENTS

1.0 Introduction

The purpose of this Implementing Standard is to consolidate the standards to be applied in the design, construction, and operation of the WTP with respect to defense in depth. This Implementing Standard also provides for tailoring of defense in depth as is appropriate to the nature and severity of the hazard and hazardous situations to which it is applied.

Section 2.0 identifies the subordinate implementing standards used in the application of the six defense in depth sub-principles of DOE/RL-96-0006. These subordinate standards are derived, in part, from various available consensus standards. In cases where no relevant consensus standard exists for a given defense in depth sub-principle, this document provides the criteria to be implemented.

Section 3.0 discusses the approach to be used in implementing defense in depth with respect to determining an adequate combination of passive and active barriers that afford protection against a postulated initiating event.

Section 4.0 provides definitions of terms used in this Implementing Standard. These definitions are derived from DOE/RL-96-0006 and consensus standards, tailored to the work and hazards of the WTP.

Section 5.0 lists the subordinate implementing standards identified in section 2.0 and describes any necessary tailoring.

Section 6.0 lists the references used in this Implementing Standard.

2.0 Standards for the Implementation of Defense in Depth Sub-Principles

The following sub-principles must be addressed in order to demonstrate compliance with the principle of defense in depth, as formulated in DOE/RL-96-0006 and DOE O 420.1A:

- Defense in depth
- Prevention
- Control
- Mitigation
- Automatic Systems
- Human Aspects
- Preparation for Emergencies

The following subsections contain the standards on application of the seven sub-principles of defense in depth from DOE/RL-96-0006. These standards will be tailored to remove obviously reactor-specific and other non-applicable criteria. In accordance with the DOE/RL-96-0004 process, further tailoring will be performed as the design develops.

The following subsections contain excerpts and extracts from several consensus standards. Where necessary to avoid the implication of misquoting, differences in wording from the cited consensus standards are identified by presenting added words in italics and by inserting double-brackets where words have been removed. Citation of a portion of a given consensus standard shall not be read to infer that other portions of the standard not specifically cited are being invoked.

2.1 Defense in Depth

"To compensate for potential human and mechanical failures, a defense-in-depth strategy should be applied to the facility commensurate with the hazards such that assured safety is vested in multiple, independent safety provisions, not one of which is to be relied upon excessively to protect the public, the workers or the environment. This strategy should be applied to the design and operation of the facility." (DOE/RL-96-0006, Section 4.1.1.1)

2.1.1 Implementing Standards

1. DOE O 420.1A, *Facility Safety* (Ref. 5.2), section 4.1.1.2, first three paragraphs only
2. DOE G 420.1-1 *Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria Guide for use with DOE O 420.1 Facility Safety*, section 2.3, except last paragraph
3. ANSI/ANS-58.9-1981, *Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems*
4. IEEE Std 379-1994, *IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems*

2.1.2 Discussion

The WTP will be designed with the objective of providing multiple layers of protection to prevent or mitigate the unintended release of radioactive materials to the environment. Defense in depth will include: siting; minimization of material at risk; the use of conservative design margins and quality assurance; the use of successive physical barriers for protection against the release of radioactivity; the provision of multiple means to control critical safety functions (those basic safety functions needed to control the processes, maintain them in a safe state, and to confine and mitigate radioactivity associated with the potential for accidents with significant [ ] radiological impact *to the public, facility workers or co-located workers*); the use of equipment and administrative controls which restrict deviations from normal operations and provide for recovery from accidents to achieve a safe condition; means to monitor accident releases required for emergency responses; and the provision of emergency *preparedness* for minimizing the effects of an accident DOE O 420.1A.

The defense-in-depth concept is integrated into the WTP design process. The application of the defense-in-depth concept to the facility design helps identify potential safety features to be included in the facility design. Consideration will be given to prevent or mitigate accident consequences from contaminating the environment, even when direct public or facility or co-located worker safety is not an issue.

Defense in depth is a safety design concept or strategy that is applied at the beginning and will be maintained throughout the facility design process. This safety design strategy is based on the premise that no one layer of protection is completely relied upon to ensure safe operation. This safety strategy provides multiple layers of protection to prevent or mitigate an unintended release of radioactive material to the environment.

Conceptually, there are three layers of defense in depth.

1. The first layer of defense consists of a well-designed facility with process design to reduce source terms, reliable SSCs that are simple to operate and maintain and resistant to degradation, and personnel well trained in operations and maintenance and committed to a strong safety culture.
2. The second layer recognizes that failures of systems and components and human failures cannot be entirely eliminated and that protective features (e.g., engineering design features and administrative controls) are required. These features are provided to ensure a return to normal operation or to bring the facility to a safe condition in the event of anticipated, but abnormal events. These features may provide automatic system response to such events or may be monitors that alert operators to the necessity of taking manual action. Such response to off-normal conditions can effectively halt the progression of events toward an accident.
3. The final layer of defense consists of conservatively designed *important to* safety SSCs to prevent or mitigate the consequences of accidents that may be caused by errors, malfunctions, or events that occur both internal and external to the facility (DOE G 420.1-1).

Implementing Standards for the following elements of defense in depth described in DOE G 420.1-1 related to safety design and construction are addressed in the sections of this document that are referenced below.

DOE G 420.1-1 Element	Discussed in Section
Siting	2.2.2
Material at risk	2.2.2
Conservative design	2.2.2
Quality assurance	2.6.2
Physical barriers	2.4.2
Critical safety functions	2.3.2
Equipment and administrative controls	2.3.2 and 2.6.1
Emergency features	2.5.2

When the single failure criterion is implemented, it is completed in accordance with ANSI/ANS-58.9-1981 for fluid systems and IEEE Std 379-1994 for electrical and instrumentation and control systems using related single failure criteria and redundancy discussion from DOE G 420.1-1, Section 5 Supplementary Design Criteria for Safety Structures, Systems and Components as additional guidance. This provides a tailored approach requiring single failure criterion protection for SDC/SDS control strategies that protect against SL-1 events and requiring single failure criterion protection be considered for SDC/SDS control strategies that protect against SL-2 events. Likewise this provides a tailored approach requiring single failure criterion for safety-class systems and components and requiring single failure criterion protection be considered for safety-significant systems and components.

The application of the single failure criterion begins with the identification of an initiating event. Initiating events are identified in the normal course of applying integrated safety management in accordance with DOE/RL-96-0004, as described in the WTP Implementing Standard for Safety Standards and Requirements Identification (i.e., SRD Vol. II, Appendix A). In evaluating the defense in depth of the WTP, single failures must be postulated in addition to the initiating event (that is the initiating event is not the single failure) (ANSI/ANS-58.9-1981). For fluid systems, during the short term, the single failure considered may be limited to an active failure. During the long term, assuming no prior failure during the short term, the limiting single failure considered can be either active or passive. Examples of passive failures are valve packing and pump seal leakage.

2.2 Prevention

"Principal emphasis should be placed on the primary means of achieving safety, which is the prevention of accidents, particularly any that could cause an unacceptable release." (DOE/RL-96-0006, Section 4.1.1.2)

2.2.1 Implementing Standards

1. DOE O 420.1A, section 4.1.1.2, first three paragraphs only
2. DOE G 420.1-1, section 2.3, except last paragraph

2.2.2 Discussion

The provision of hazard elimination and protection shall be optimized by measures such as the choice of siting, proven conservative design and construction, a robust start-up testing program, operating requirements (i.e., clear definition of normal and abnormal operating conditions and maintenance activities).

> **Siting**. The WTP site location will reduce the need to provide design measures to alleviate potentially hazardous conditions or to protect surrounding populations (for example, consideration of ground instability, river flooding, and hazards due to nearby industrial installations or activities) (DOE G 420.1-1).
>
> **Material at Risk**. The WTP and its process design and administrative controls will minimize and control inventories of radioactive materials and their forms (DOE G 420.1-1).
>
> **Conservative Design**. The WTP design will include conservative margins that allow flexibility of operations and maximize the time before requiring corrective actions. These margins will also take into consideration the potential degradation of elements and operational errors (DOE G 420.1-1).

The site for the facility has been established by DOE. Aspects of siting that remain for consideration include:

1 The risk that the site presents to the facility in terms of natural phenomena and nearby industry and transportation, and

2 The risk that the facility presents to the nearby environment, co-located workers, and the public.

Defense in depth for protection against NPH events is achieved by:

1. The selection of NPH loadings of SRD Safety Criterion 4.1-3 that have a low frequency of occurrence in the lifetime of the facility with the most severe events having the lowest frequency of occurrence, and
2. The selection of design, fabrication, and construction standards that provide a significant margin to failure should the NPH loading be experienced.

Protection against accidents at nearby industry and transportation locations is addressed by conservative analyses of radiological and chemical release, overpressure, and physical impact events related to these facilities.

The vitrification project does not have control over the environment or population (co-located worker and public) outside the controlled area. However, all of the sub-principles of defense in depth discussed in Section 2.0 provide for protection of the environment, co-located worker, and public against the uncontrolled release of chemical and radiological materials from the facility.

The design shall address all identified hazards and hazardous situations and pursue methods for their prevention. The preferred means of prevention is to eliminate or reduce the severity of the hazard itself. According to the Implementation Guide on nonreactor facility safety, one objective of prevention as an element of defense in depth is to apply facility and process design and administrative controls to minimize and control inventories of radioactive materials and their forms (that is, minimize the material at risk) (DOE G 420.1-1).

Elimination or reduction of the hazard can be achieved by substituting less hazardous materials in processing, limiting the inventory of the material, etc. The design process must provide evidence through documentation that this option was considered and implemented to the maximum extent practicable. Where the hazard itself cannot be eliminated or reduced, controls shall be provided to reduce the likelihood of the hazard manifesting itself into an accident. Where hazard elimination is not practicable, passive features are to be employed, since they are simple and have a high degree of reliability. Where this is not practicable, active protection will be proposed that has a degree of reliability and confidence commensurate with the potential hazard severity.

Conservatism in design is achieved in part by requiring a significant margin between the design limit and the ultimate failure point of a SSC. Conservatism in design is also accomplished by giving preference to passive over active components, material selection, keeping systems as simple in their operation and maintenance as possible, including provisions for corrosion and erosion, prevention, and the mitigation of mis-operation of systems and components (e.g., by the use of interlocks), and redundancy and diversity to accommodate system and component failures.

2.3 Control

"Normal operation, including anticipated operational occurrences, maintenance and testing, should be controlled so that facility and system variables remain within their operating ranges and the frequency of demands placed on structures, systems and components important to safety is small." (DOE/RL-96-0006, Section 4.1.1.3)

2.3.1 Implementing Standards

1. DOE O 420.1A, section 4.1.1.2, first three paragraphs only
2. DOE G 420.1-1, section 2.3, except last paragraph
3. ISA-S84.01-1996, Application of Safety Instrumented Systems for the Process Industries

2.3.2 Discussion

The DOE Implementing Guidance for nonreactor facility safety provides two criteria related to the defense in depth sub-principle of control:

Critical safety functions. Design to provide multiple ways for safety functions to control processes, to maintain processes in a safe state, and to confine radioactivity when accidents could have the potential for significant [ ] radiological impact *to the public, facility workers or co-located workers* (DOE G 420.1-1).

Equipment and administrative controls. Include features to control process variables to values within safe conditions, to alert operating personnel of an approach toward conservative process limits, to allow timely detection of failure or malfunction of critical equipment, and to allow for the imposition of administrative controls assumed in the hazard analysis, and/or accident analysis (DOE G 420.1-1).

Normal operations, which include anticipated operational occurrences and maintenance and testing activities, shall be controlled so that facility and system parameters remain within their specified operating ranges and that the frequency of demands placed on SSCs for hazard prevention and mitigation is small.

This will be achieved by the choice of design that will:

1. Control key operating parameters such that facility operations remain within the safe operating envelope. Key operating parameters are those that define how the plant will be operated safely.
2. Maintain the safe operating envelope (e.g., a wide variation in operation conditions can be tolerated without entering into a potentially unsafe region).
3. Ensure that any failure mode would not move the facility or process toward a potentially unsafe region (i.e., fail to safe state).
4. Provide instrumentation and control features (e.g., temperature, pressure, radiation monitoring) which will warn of reduced margins of safety and, where appropriate, automatically return the process into the designated safe operating regime.
5. Achieve independence between SSCs credited for control of normal facility operations and those credited for prevention and mitigation of potential hazards.

For example, assume that the normal operating temperature range in an ion exchange column is set at 30 - 50 °C and that column temperatures above 80 °C lead to enhanced resin degradation and a potential explosion hazard. Engineered controls for maintaining that temperature within the normal operating limits (e.g., temperature control system) will be independent of that which would alert the operator and perform a preventative action (e.g., shut down process, increase cooling, etc.) in order that the hazard could not occur.

2.4 Mitigation

"The facility should be designed to retain the radioactive material through a conservatively designed confinement system for the entire range of events considered in the design basis. The confinement system should protect the workplace and the environment." (DOE/RL-96-0006, Section 4.1.1.4)

2.4.1 Implementing Standards

1. DOE O 420.1A, section 4.1.1.2, first three paragraphs only
2. DOE G 420.1-1, section 2.3, except last paragraph
3. Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02, see below)

2.4.2 Discussion

Mitigation is implemented to ensure reduction of consequences from potential hazards and hazardous situations such that the applicable exposure standards are satisfied. One method of achieving this element of defense in depth is to ensure that suitable confinement of radioactive and hazardous material is maintained throughout normal operation and credible accident conditions. Confinement will be achieved by physical barriers and by other SSCs that either assure integrity of the physical barriers or minimize the quantity and characteristics of any hazardous material potentially releasable.

DOE Order 420.1A, requires:

> "All nuclear facilities with uncontained radioactive materials (as opposed to material contained within drums, grout and vitrified materials) shall have means to confine them. Such confinement will act to minimize the spread of radioactive materials and the release of radioactive materials in facility effluents during normal operations and potential accidents. For a specific nuclear facility, the number and arrangement of confinement barriers and their required characteristics shall be determined on a case-by-case basis. Factors that shall be considered in confinement system design shall include type, quantity, form, and conditions for dispersing the material. Engineering evaluations, trade-offs, and experience shall be used to develop practical designs that achieve confinement system objectives. The adequacy of confinement systems to effectively perform the required functions shall be documented and accepted through the Safety Analysis Report." (DOE G 420.1-1)

DOE G 420.1-1 defines confinement barriers to include primary confinement and secondary confinement. "Primary confinement provides confinement of hazardous material to the vicinity of its processing -- typically by means of piping, tanks, glove boxes, encapsulating material, etc., along with any offgas systems that control effluent from the primary confinement. As such, primary confinement addresses the preventive sub-principle of defense in depth, as well as mitigation. Secondary confinement consists of a cell or enclosure surrounding the process material or equipment along with any associated ventilation exhaust systems from the enclosed area." [ ] (DOE G 420.1-1)

The WTP will provide physical barriers to confine radioactive material and thereby prevent uncontrolled releases. In general, multiple physical barriers - i.e., primary and secondary confinement - will be provided, especially for the most severe hazards and hazardous situations. The provision of multiple physical barriers will be tailored to the work and associated hazards, as discussed in section 3.0.

DOE G 420.1-1 suggests several industry consensus codes and standards for the design and construction of the SSCs comprising confinement, as follows: structures - subsection 5.2.1, ventilation systems - subsection 5.2.2.1, and process equipment - subsection 5.2.2.2. The specific standards for SSCs that implement mitigation with respect to SSCs comprising confinement are contained in the following Safety Criteria from the Safety Requirements Document Volume II:

- Structures - SC 4.1-2
- Ventilation systems - SC 4.4-3
- Process equipment - SC 1.0-5, 4.2-1 through 4.2-3

2.5 Automatic Systems

"Automatic systems should be provided that would place and maintain the facility in a safe state and limit the potential spread of radioactive materials when operating conditions exceed predetermined safety setpoints." (DOE/RL-96-0006, Section 4.1.1.5)

2.5.1 Implementing Standards

1. ISA-S84.01-1996, *Application of Safety Instrumented Systems for the Process Industries*
2. ANSI/ANS-58.8-1994, *Time Response Design Criteria for Safety-Related Operator Actions*

2.5.2 Discussion

Automatic systems shall be provided to prevent the facility from entering into or remaining within an unsafe regime that may lead to the potential for radioactive or hazardous material release to facility and co-located workers, the public, or the environment, except as discussed below. The definition of the boundaries between safe and unsafe regimes will be determined as a result of detailed facility design, start-up, and testing activities. This will allow the derivation of the predetermined setpoints for safe facility operations. Automatic systems will be part of the overall suite of SSCs provided as part of the hazard control strategy. The determination of the need for automatic systems will be assessed as part of the determination of the overall hazards control strategy.

Means shall be provided to automatically initiate and control all protective actions except as justified below.

Credit for operator action may be permissible only if safety analysis demonstrates that the total time interval required to perform the operator action exceeds the time at which the limiting design requirement would be reached without operator action, in accordance with the methodology of ANSI/ANS-58.8-1994.

2.6 Human Aspects

"The human aspects of defense in depth should include a design for human factors, a quality assurance program, administrative controls, internal safety reviews, operating limits (Technical Safety Requirements), worker qualification and training, and the establishment of a safety/quality program." (DOE/RL-96-0006, Section 4.1.1.6)

2.6.1 Implementing Standards

1. IEEE Std 1023-1988, *IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations.*
2. Implementing standards for the quality assurance program, administrative controls, internal safety reviews, operating limits (Technical Safety Requirements), worker qualification and training, and the establishment of a safety/quality program are contained in the Safety Requirements Document Volume II, as discussed below.

2.6.2 Discussion

Design for Human Factors

The design shall apply human factors engineering (HFE) to address the ergonomic requirements of facility operations and maintenance of the WTP. The DOE nonreactor Implementation Guide recommends that the following human factor elements be considered: equipment labeling, workplace environment (temperature and humidity, lighting, noise, vibration, and aesthetics), human dimensions, operating panels and controls, component arrangement, warning and annunciator systems, and communication systems (DOE G 420.1-1).

The WTP design engineers, in consultation with operators, will apply these HFE elements in the design of important to safety SSCs to ensure that operational preferences are implemented. Human factors engineering specialists will provide support in the application of HFE.

Human factors engineering shall be conducted in accordance with IEEE Std 1023-1988, as discussed below. Selection of this subordinate standard conforms with DOE G 420.1-1.

IEEE Std 1023-1988 was developed specifically for nuclear power generating stations. Therefore, this subordinate standard will be tailored to the work and hazards of the WTP as follows. The formal HFE process described in subsection 6.1.1 of IEEE Std 1023-1988 will be applied to the evaluation of hazards whose consequences fall into the two highest severity levels - SL-1 and SL-2 (see in SRD Volume II, Appendix A, section 4.3.1).

Although the structured HFE program outlined in subsection 6.1.1 of IEEE Std 1023-1988 will not be implemented for SL-3 and SL-4 events, the general HFE elements will be considered for all ITS SSCs, as committed above.

Similarly, formal consideration of the HFE techniques and methodologies recommended in section 5 of IEEE Std 1023-1988 will be undertaken for hazards of severity levels SL-1 and SL-2. Certain of these techniques and methodologies may be utilized in the evaluation of SL-3 and SL-4 events in the context of the normal design and hazard assessment and control effort, as part of the integrated safety management process.

Quality Assurance Program

The *Safety Requirements Document* Safety Criterion 7.3-1 requires the WTP contractor to establish and implement a quality assurance program compliant with 10 CFR 830, Subpart A. This program is being implemented in accordance with the *Quality Assurance Manual* (QAM) (24590-WTP-QAM-QA-01-001).

The QAM applies specifically to work performed on or for the WTP. The QAM is in conformance with 10 CFR 830, Subpart A and with the top-level principles stated in DOE/RL-96-0006 .

Administrative Controls

Administrative controls include features to control process variables to values within normal and safe conditions, to monitor equipment status, to alert operating personnel of an approach toward conservative process limits, to allow timely detection of failure or malfunction of critical equipment, and to allow for the imposition of administrative controls assumed in the hazard analysis, and/or accident analysis (DOE G 420.1-1).

The primary means of implementing defense in depth is through the provision of multiple physical barriers that maintain confinement. The output of the design process, through which hazards and hazardous situations are identified, control strategies implemented and standards defined will be a set of SSCs that contribute to defense in depth. SSCs so identified will always be backed up by administrative controls such as procedures. Administrative controls that afford a measure of defense in depth will be developed prior to facility operations. For the purpose of protecting the public and co-located worker, administrative controls alone shall not be relied on for the implementation of defense in depth. Administrative controls alone may be credited as the controls that protect facility workers, when appropriate. In such cases, defense in depth is provided through other human aspects, such as worker qualification and training.

Internal Safety Reviews

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Safety Criterion 7.1-3, requires that the WTP contractor establish a safety framework and specifies requirements for the Internal Safety Oversight program consistent with Top-Level Principle 4.4.1, "Safety Review Organization". BNI has established a WTP Project Safety Committee (PSC) to provide an independent, interdisciplinary evaluation of matters related to nuclear, radiological, and process safety.

Operating Limits (Technical Safety Requirements)

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Safety Criterion 9.2-1, commits the WTP contractor to prepare, submit for approval, and operate the facility in accordance with Technical Safety Requirements (TSRs). SCs 9.2-2 through 9.2-5 provide the safety criteria for the bases and contents, updating, submission for regulatory approval, and maintenance of TSRs.

As part of hazard evaluation, the role of the operator in the development of a potential hazard will be identified and reliability assessed. Human factors specialists in the multidisciplinary team will support this evaluation. The results of the assessment will be incorporated into administrative controls such as operating procedures and TSRs.

Worker Qualification and Training

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), section 7.2, commits the WTP contractor to establish and implement a training program. Consistent with Top-Level Principles 4.3.4.1, "Personnel Training", 4.3.4.2, "Training Programs", and 5.2.4, "Process Safety - Training," SRD Volume II, section 7 requires that the program address:

- continual training - SC 7.2-1 and 7.2-2
- qualification of personnel - SC 7.3-1
- establishment of written procedures/instructions - SC 7.2-3

Establishment of a Safety/Quality Program

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Safety Criteria 1.0-1, requires the use of a comprehensive safety management program consistent with Top-Level Principle 5.1.1, "Process Safety Management", and 5.1.2, "Process Safety Objective". Safety Criterion 7.1-3 requires a safety framework be established to implement this Program consistent with Top-Level Principle 4.1.4.1, "Safety/Quality Culture".

Establishment of a Quality Program is discussed above under the heading, "Quality Assurance Program".

2.7 Preparation for Emergencies

"Non-reactor nuclear facilities shall be designed with the objective of providing multiple layers of protection to prevent or mitigate the unintended release of radioactive materials to the environment. Defense in depth shall include: siting, minimization of material at risk, the use of conservative design margins and quality assurance; the use of successive physical barriers for protection against the release of radioactivity; the provision of multiple means to ensure critical safety functions (those basic safety functions needed to control processes, maintain them in a safe state, and to confine and mitigate radioactivity associated with the potential for accidents with significant radiological impact); the use of equipment and administrative controls which restrict deviations from normal operations and provide for recovery from accidents to achieve a safe condition; means to monitor accident releases for emergency responses; and the provision of emergency plans for minimizing the effects of an accident."
(DOE O 420.1A, Section 4.1.1.2, first paragraph)

2.7.1 Implementing Standards

1. Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02, see below)

2.7.2 Discussion

Accident Release Monitors

The WTP will provide the capability to monitor accident releases as necessary to support emergency responses.

Emergency Plan

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Section 7.8 requires an emergency preparedness plan.

3.0 Determination of SSCs for the Implementation of Defense in Depth

The standards for prevention, control, and human aspects in sections 2.2, 2.3, and 2.6 are primarily concerned with defense in depth sub-principles that minimize the potential of hazard initiation. In evaluating accidents that are postulated to occur despite implementation of preventive, control and human aspects, the sub-principles of mitigation and automatic systems must be considered.

The Implementing Standard for Safety Standards and Requirements Identification, SRD Volume II, Appendix A, describes the process by which hazards and hazardous situations are identified and evaluated to determine hazard control strategies. Use of SRD Appendix A with this Appendix B Implementing Standard for Defense in Depth ensures that the defense in depth sub-principles are accounted for in the process of determining hazard control strategies. That process will identify SSCs that contribute to defense in depth as part of their safety function. The administrative controls that back up these SSCs will be developed prior to the introduction of hazardous materials into the facility.

In addition to the identification of defense in depth SSCs through implementation of SRD Volume II, Appendices A and B, the requirement to satisfy the accident risk goals of SRD Safety Criteria 1.0-2 may require the identification of additional accident prevention or mitigation SSCs.

3.1 Radiological Release Events

Table 1 is the standard for implementing defense in depth by SSCs as part of the preferred hazard control strategy; it defines the minimum number of controls and associated engineering requirements for the control of radiological release hazards of a particular severity.

Table 1 will be used in conjunction with the guidance in section 2.0 to ensure that the preferred hazard control solution addresses the strategies that protect the public and co-located workers from the uncontrolled release of radiological materials; such SSCs will always be backed up by the human aspects of defense in depth discussed in section 2.6.

The table lists the number and attributes of the physical barriers. Consistent with the defense in depth sub-principles in section 2.0, the preferred hazard control strategy should emphasize passive SSCs over active SSCs.

Tying the number of physical barriers to the hazard's severity level contributes to achieving defense in depth in accordance with the tailored approach mandated by RL/REG 98-17, "Regulatory Unit Position on Tailoring for Safety."

1st Column - SL (Severity Level)

Determination of hazard severity level is based on an assessment of unmitigated consequences as discussed in SRD Volume II, Appendix A, Section 4.3.1. Severity levels are defined as SL-1 to SL-4, with SL-1 having the highest consequences.

2nd Column - Control Options for Implementation of Defense in Depth

A graded approach is reflected in the configuration requirements specified for each hazard severity level and receptor. The requirements are more stringent for defense in depth implementation for hazards of greater severity than for those of lesser severity and likewise greater for the public receptor relative to the worker receptor. Events with SL-2 and SL-3 consequences to the public potentially have SL-1 consequences to the co-located worker. Events with SL-4 consequences to the public potentially have SL-1 or SL-2 consequences to the co-located worker. The most stringent of these requirements shall apply.

Radiological release events that affect only the facility worker are qualitatively assessed in order to determine if additional barriers (i.e., SSCs, adminstrative controls, or both) are needed to provide appropriate defense in depth. Protection of the public is predominant in safety design; protection of workers is no less important. However, the degree of protection for facility workers achievable by SSCs is limited. Other factors such as disciplined conduct of operations, training, and safety management programs are no less important in assuring worker safety (DOE G 420.1-1).

Implementation of defense in depth requires that the single failure criterion be applied in a tailored fashion. The single failure criterion is discussed in section 2.1.

In addition to the single failure criteria diversity may also be implemented in the control strategy where hazards assessment reveals a common mode failure concern (see the Implementing Standard for Safety Standards and Requirements Identification, SRD Vol. II, Appendix A).

Implementation of defense in depth also requires that the provision of physical barriers be applied in a tailored fashion as noted in Table 1. For SL-1 and SL-2, two or more independent physical barriers are required. For SL-3, at least one physical barrier shall be provided, and two or more independent physical barriers shall be considered; that is, an objective assessment must be performed to determine the extent to which physical barriers will be incorporated by the design. The results and basis of this assessment shall be documented. Such documentation shall be retrievable and can be in various forms such as engineering studies, meeting minutes, reports, or internal memoranda.

The graded approach is also reflected in the degree of confidence required commensurate with the hazard severity. The confidence is based on the standards and other attributes applicable to the particular control strategy. The Implementing Standard for Safety Standards and Requirements Identification describes selection of standards and other attributes applicable to control strategies.

Table 1. Implementation of Defense in Depth by SSCs (Safety Design Class/Safety Design Significant/Risk Reduction Class).

Severity Level	Control Options for Implementation of Defense in Depth
SL-1	Two or more independent physical barriers. The single failure criterion shall be applied as appropriate.
SL-2	Two or more independent physical barriers. The single failure criterion shall be considered.
SL-3	At least one physical barrier shall be provided. Two or more independent physical barriers shall be considered.
SL-4	Physical design features and/or administrative controls per 10 CFR 835.1001
Physical design features or administrative controls alone may be credited as the controls that protect facility workers, when appropriate. Timely evacuation from the vicinity of the hazard is considered to be an administrative control. Physical barriers are not required for those events that are prevented.	

Table 1A Implementation of Defense in Depth by SSCs (Safety-Class, Safety-Significant, or Additional-Protection Class)

Severity Level	Control Options for Implementation of Defense in Depth for Co-located Worker and Public*
SL-1	Two or more independent physical barriers.
SL-2	Two or more independent physical barriers.
SL-3	At least one physical barrier shall be provided. Two or more independent physical barriers shall be considered.
SL-4	Physical design features and/or administrative controls per 10 CFR 835.1001
Facility Worker Consequence Ranking	***Control Options for Implementation of Defense in Depth for Facility Worker***
High	At least one barrier shall be assigned to prevent or mitigate the impacts to the facility worker: • If an administrative control barrier is selected, it must be developed into an SCR and TSR that capture the specific safety function related to the hazard • If a barrier is selected that already has a safety function for protecting the co-located worker or public, the worker safety function shall be explicitly stated for that barrier

Moderate or Low	As a minimum, the ISM team shall document the barriers or safety management programs (e.g., radiation protection, hazardous material protection, maintenance, procedures, training) that are present to prevent or mitigate impacts to facility workers from the hazard. These barriers or safety management programs shall be documented in the ISM meeting minutes and SIPD database.
Standard Industrial	Nuclear and chemical industrial hazards are adequately controlled by adherence to safety programs that implement worker safety requirements. These hazards are not required to be documented in SIPD.

* Hazard control strategies that do not meet these minimum requirements shall be approved using the Contract-approved methodology for making such changes.

3.2 Direct Radiation Events

3.2.1 Direct Radiation Events for SSCs Classification as SDC, SDS or RRC

Because of the distances involved, direct radiation is primarily a hazard to the facility worker as opposed to the co-located worker or the public. Direct radiation hazards usually involve:

1. Accidents that result in a release of radiological material
2. Inadvertent facility worker entry into an area with a high radiation field.

Mitigation of the first type (accidents involving a radiological release) is usually accomplished by the use of passive shield walls. Prevention of the second type (entry into a high radiation field) usually involves the use of engineered and administrative controls to prevent the entry into areas with a high radiation field.

Implementation of defense in depth by SSC for direct radiation events begins in a manner similar to that used for radiological releases; that is, by the assignment of severity levels based upon unmitigated consequences.

Table 2 is the standard for implementing defense in depth by SSCs as part of the preferred hazard control strategy related to the prevention and mitigation of direct radiation accidents. The basic description of the first and third columns is the same as that provided in section 3.1 for accidents involving radiological releases.

Table 2. Implementation of Defense in Depth by SSC for Direct Radiation Hazards.

Severity Level (SL)	Control Options for Implementation of Defense in Depth
SL-1	One passive physical barrier that is not challenged by the event; two independent barriers if the first barrier might be challenged by the event or is not totally passive.

Severity Level (SL)	Control Options for Implementation of Defense in Depth
SL-2	One passive physical barrier that is not challenged by the event; two independent barriers if the first barrier might be challenged by the event or is not totally passive.
SL-3	One physical barrier.
SL-4	One barrier (physical or administrative).
Administrative controls alone may be credited as the controls that protect facility workers, when appropriate. Timely evacuation from the vicinity of the hazard is considered to be an administrative control. Physical barriers are not required for those events that are prevented.	

The unmitigated event frequency must also be calculated for passive SSCs that might be challenged by the event, however, where passive barriers are provided and the barriers would not be challenged by the event (e.g., insignificant pressurization of a cell relative to its inherent strength) it is not necessary to estimate probability of failure to determine the unmitigated event frequency.

3.2.2 Direct Radiation Events for SSCs Classified as SC, SS, or APC

Because of the distances involved, direct radiation is primarily a hazard to the facility worker as opposed to the co-located worker or the public. The facility worker is protected from direct radiation exposure during normal operation by the design of passive shield walls to satisfy the requirements of 10 CFR 835. Accidental exposure to direct radiation hazards usually involve:

1. Accidents that result in a release of radiological material
2. Accidents that result in a loss of shielding, or
3. Inadvertent facility worker entry into an area with a high radiation field.

Accidents of the first type are addressed in section 3.1.

Accidents of the second type are unlikely because radiation shields are generally massive, passive barriers that are not readily degraded or removed. The consequences of such accidents are mitigated by evacuation.

Accidents of the third type are prevented by a combination of administrative and physical barriers in accordance with the requirements of 10 CFR 835.

3.3 Chemical Release

The potential consequences of hazardous chemicals shall also be assessed. The assessment shall consider both the inherent hazard of the chemical itself, and the potential for the chemical hazard to initiate or exacerbate a radiological hazard.

As many of the chemical hazards of the vitrification facility are not unique to the facility, the selection of preferred hazard control strategies includes identification of what has been required and accepted as engineered prevention and mitigation features for industrial plants with a similar chemical hazard. The chemical hazard for the vitrification facility is also reviewed to determine if it has a chemical risk that is somewhat unique to the facility. When such a case is identified, consideration is given to additional (or augmented) accident prevention and/or mitigation engineered features.

Additional detail on the selection of preferred hazard control strategies for chemical hazards and hazardous situations is provided in the SRD Volume II, Appendix A, "Implementing Standard for Safety Standards and Requirements Identification".

4.0 Definitions

Definitions of the following terms were obtained from the referenced consensus standards. Minor wording differences among multiple references are ignored. In some cases, the definition of a term given in the referenced consensus standard has been tailored to the relative risks of the WTP and its anticipated associated hazards. Other wording differences in the definitions below from the cited consensus standards have been made to preserve consistency with terminology in other WTP safety documentation. Such differences are identified by presenting added words in Italics and by inserting double-brackets where words have been removed. Citation of a definition from a given consensus standard shall not be read to infer that other portions of the standard not specifically cited are being invoked.

Active component [SSC]. A component in which mechanical movement must occur to accomplish the [ ] safety function of the component (ANSI/ANS-51.1-1983 and ANSI/ANS-52.1-1983)

Active failure. A malfunction, excluding passive failures, of a component that relies on mechanical movement to complete its intended [ ] safety function upon demand

Examples of active failures include the failure of a valve or check valve to move to its correct position, or the failure of a pump, fan, or diesel generator to start.

Spurious action of a powered component originating within its actuation or control system shall be regarded as an active failure unless the specific design features or operating restrictions preclude such spurious action. An example is the unintended energization of a powered valve to open or close (ANSI/ANS-51.1-1983, ANSI/ANS-52.1-1983, and ANSI/ANS-58.9-1981).

Administrative controls. Provisions relating to organization and management, procedures, record keeping, assessment, and reporting necessary to ensure safe operation of the facility.

Barrier. A control (typically part of a control set or strategy) that is preventing or mitigating either: (1) the release of radioactive or hazardous material to the facility or co-located worker, public, or the environment; or (2) the exposure at the facility or co-located worker or the public to sources of direct radiation. This control can be an SSC that provides a physical barrier (e.g. vessel, confinement, shielding, and filtration) or a physical design feature that supplements the physical barrier such as equipment or emergency features (e.g., process controls, detectors, alarms, and monitors) or an administrative control (e.g., training and procedures).

Common cause failure. Dependent failures that are caused by a condition external to a system or set of components that make system or multiple component failures more probable than multiple independent failures (DOE/RL-96-0006).

Common mode failure. Dependent failures caused by susceptibilities inherent in certain systems or components that make their failures more probable than multiple independent failures due to those components having the same design or design conditions that would result in the same level of degradation (DOE/RL-96-0006).

Confinement barrier. Physical barrier that prevents or mitigates the release of radioactive or hazardous material to the worker, public or the environment. The DOE nonreactor facility safety Implementation Guide identifies three kinds of confinement barriers - primary confinement, secondary confinement, and tertiary confinement (DOE G 420.1-1).

Control strategy. A set of generally-described provisions (barriers, dilution/dispersal, physical limitations on material quantities, administrative material controls, confinement, ventilation of flammable gas, etc.) and/or approaches (defense in depth, use of passive features, prevention, mitigation, etc.) which are intended to assure adequate control of a specific hazard and associated accidents in the context of the work (DOE/RL-96-0006).

Defense in depth. The fundamental principle underlying the safety technology of the facility centered on several levels of protection including successive barriers preventing the release of radioactive materials to the workplace or the environment. Human aspects of defense in depth are considered to protect the integrity of the barriers, such as quality assurance, administrative controls, safety reviews, operating limits, personnel qualifications and training and safety program. Design provisions including both those for normal facility systems and those for systems important to safety help to: 1) prevent undue challenges to the integrity of the physical barriers; 2) prevent failure of a barrier if challenged; 3) where it exists, prevent consequential damage to multiple barriers in series; and 4) mitigate the consequences of accidents. Defense in depth helps to assure that the basic safety functions are preserved and that radioactive materials do not reach the worker, public or the environment (DOE/RL-96-0006).

Dependent Failures. In general, dependent failures are defined as events in which the probability of each failure is dependent upon the occurrence of other failures (Modarres 1993).

Design Basis Events. Postulated events providing bounding conditions for establishing the performance requirements of structures, systems and components that are necessary to: (1) ensure the integrity of the safety boundaries protecting the worker; (2) place and maintain the facility in a safe state indefinitely; or (3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design Basis Events also establish the performance requirements of the structures, systems, and components whose failure under Design Basis Event conditions could adversely affect any of the above functions (DOE/RL-96-0006).

Detectable failures. [The following definition is considered to be specific to electrical, instrumentation and control systems.]

Failures that can be identified through periodic testing or can be revealed by alarm or anomalous indication (IEEE Std 379-1994).

Diversity. Use of different technologies, equipment, or design methods to perform a common function with the intent to minimize common cause failures (ISA-S84.01-1996).

Engineered feature. A structure, system or component that contributes to the safe operation of the facility (24590-WTP-ISMP-ESH-01-001).

Event. A condition that deviates from normal operation, i.e., an initiating occurrence plus single failure or coincident occurrence combination (ANSI/ANS-51.1-1983 and ANSI/ANS-52.1-1983).

External Event. An event external to the WTP caused by (1) a natural hazard (e.g., earthquake, flood, lightning, or range fire) or (2) a human-induced event (e.g., transportation or nearby industrial activity).

Human factors engineering (HFE). An interdisciplinary science and technology concerned with the process of designing for human use (IEEE Std 1023-1988).

Important to Safety. Structures, systems and components that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation).

This definition includes not only those structures, systems and components that perform safety functions and traditionally have been classified as safety class, safety-related or safety grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems and components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of important to safety, i.e., safety-related may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems and components should be considered for inclusion within this definition (DOE/RL-96-0006). The WTP has two classification schemes for Important to Safety SSCs. Both of the classification schemes are divided into three separate categories and are defined to Safety Criteria 1.0-6.

Independence. The state in which there is no mechanism by which any single design basis event, such as a flood, can cause redundant equipment to be inoperable (IEEE Std 384-1992).

Initiating occurrence/event. A single occurrence and its consequential effects that place the plant or some portion of the plant in an off-normal condition. An initiating occurrence/event is not the single failure defined elsewhere herein. An initiating occurrence can be *an internal event or an external event* (ANSI/ANS-51.1-1983, ANSI/ANS-52.1-1983, and ANSI/ANS-58.9-1981).

The first event in an event sequence. Can result in an accident unless engineered protection systems or human actions intervene to prevent or mitigate the accident (AIChE).

Internal Event. An occurrence related to structure, system, and component performance or human action, or an occurrence external to the system but within the WTP that causes upset of a structure, system, or component.

Limiting design requirements. The limiting value of a design parameter that ensures that the consequences of any event do not result in:

- Violation of plant nuclear safety criteria, including off-site radiological dose criteria, or
- Unacceptable degradation of plant components that are required to mitigate the consequences of an event.

(A single event may have more than one limiting design requirement. [ANSI/ANS-58.8-1994])

Long term. *For fluid systems*, the long term is defined as that period of *important to safety* fluid system operation following the short term during which the safety function of the system is required (ANSI/ANS-58.9-1981).

Passive component. A component that is not an active component (ANSI/ANS-51.1-1983, and ANSI/ANS-52.1-1983).

Passive failure. The blockage of a process flow path or failure of a component to maintain its structural integrity or stability, such that it cannot provide its intended [ ] safety function upon demand (ANSI/ANS-51.1-1983, ANSI/ANS-52.1-1983, and ANSI/ANS-58.9-1981).

Primary confinement. Provides confinement of hazardous material to the vicinity of its processing. This confinement is typically provided by piping, tanks, glove boxes, encapsulating material, and the like, along with any offgas systems that control effluent from the primary confinement (DOE G 420.1-1).

Redundant equipment or system. A system or component that duplicates the essential functions of another system or component to the extent that either may perform the required function, regardless of the state of operation or failure of the other (IEEE Std 379-1994 and IEEE Std 384-1992).

Safety function. Any function that is necessary to ensure: 1) the integrity of the boundaries retaining the radioactive materials; 2) the capability to place and maintain the facility in a safe state; or 3) the capability to prevent or mitigate the consequences of facility conditions that could result in radiological exposures to the general public or workers in excess of appropriate limits (DOE/RL-96-0006).

Secondary confinement. Consists of a cell or enclosure surrounding the process material or equipment along with any associated ventilation exhaust systems from the enclosed area. Except in the case of housing glove-box operations, the area inside this barrier is usually unoccupied (e.g., canyons, hot cells); it provides protection for operating personnel (DOE G 420.1-1).

Shall, should and may. The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation (ANSI/ANS-51.1-1983, ANSI/ANS-52-1-1983, and ANSI/ANS-58.9-1981).

The word "shall" denotes actions that must be performed... The word "should" is used to indicate recommended practice (DOE G 420.1-1, based on DOE-STD-1075-94).

Shall [be] consider[ed]. An objective assessment must be performed to determine the extent to which the single failure criterion will be incorporated into or be satisfied by design. The results and basis of this assessment shall be documented. Such documentation shall be retrievable and can be in the form of engineering studies, meeting minutes, reports, internal memoranda, etc. (DOE O 6430.1A).

Short term. *For fluid systems*, the short term is defined as that period of operation up to 24 hours following an initiating event [ ] (ANSI/ANS-58.9).

Single failure. A random failure and its consequential effects, in addition to an initiating occurrence, that result in the loss of capability of a component to perform its intended [ ] safety function(s) (ANSI/ANS-51.1-1983, and ANSI/ANS-52.1-1983).

Single failure criterion. [Two definitions are provided below. The following definition applies to fluid (i.e., liquid and gas) systems.]

> Fluid [ ] systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly), nor (2) a single failure of any passive component (assuming active components function properly) results in a loss of the capability of the system to perform its [ ] safety function (ANSI/ANS-51.1-1983, and ANSI/ANS-52.1-1983).

[The following statement of the "single failure criterion" applies to electrical and instrumentation and control systems.]

> When required, the *important to* safety systems shall perform all required safety functions for a design basis event in the presence of the following:
>
> 1. Any single detectable failure within the *important to* safety systems concurrent with all identifiable but non-detectable failures
> 2. All failures caused by the single failure
> 3. All failures and spurious system actions that cause, or are caused by, the design basis event requiring the safety function
>
> The single failure could occur prior to, or at any time during, the design basis event for which the *important to* safety system is required to function (IEEE Std 379).

5.0 Tailoring of Consensus Standards Used in the Implementing Standard for Defense in Depth

The following subsections summarize the WTP contractor's tailoring of the consensus standards invoked by this Implementing Standard for Defense in Depth.

5.1 DOE O 420.1A, Facility Safety (Ref. 5.2)

Terminology

- Section 4.1.1.2, 1st paragraph, last sentence: Phrase "...workers, including those at adjacent facilities..." is interpreted for WTP to mean "...workers and collocated workers..."

Applicability

- The only portion of DOE O 420.1 that is being invoked by this Implementing Standard for Defense in Depth is section 4.1.1.2, the first three paragraphs.

5.2 DOE G 420.1-1 Nonreactor Nuclear Safety Criteria and Explosives Safety Criteria Guide for use with DOE O 420.1, Facility Safety

Terminology

- By virtue of cross-references within the DOE Guide 420.1-1, reference is made to "safety class" and "safety significant" SSCs. For the purposes of this guide, the WTP project uses the terms "safety design class and safety design significant", which encompass both "safety class" and "safety significant".
- "Critical safety function" in the DOE Guide 420.1-1 is interpreted to more broadly read "...significant public, worker and co-located worker impact".

Applicability

- The only portion of the DOE G 420.1-1 that is being invoked by this Implementing Standard for Defense in Depth is section 2.3, except the last paragraph.
- Section 2.3 of the DOE G contains internal cross-references to subsections 5.2.1, 5.2.2.1 and 5.2.2.2, which list typical codes for structures, ventilation systems, and process equipment that provide a confinement function. Section 2.4.2 of this Implementing Standard lists the SRD Safety Criteria that will be applied to SSCs comprising confinement.
- Section 2.3 of the DOE G contains an internal cross-reference to subsection 5.2.1, which further cites section 4.4 of DOE O 420.1A and section 3.3 of the DOE G for criteria for natural phenomena hazards (NPH). For the WTP, NPH criteria are provided in SRD Safety Criterion SC 4.1-3.

5.3 ANSI/ANS-58.8-1994, Time Response Design Criteria for Safety-Related Operator Actions

Terminology

- "Safety-related function" for the purposes of implementation of this standard is interpreted to mean "safety function" as defined in DOE/RL-96-0006, Rev 2, performed by SDC, SDS, safety-class or safety significant SSCs.

Non-Applicability

- Assumption (1) of section 1.3 does not apply. Single failure criteria for the WTP project are given in the consensus standards invoked and tailored by this Implementing Standard (ANSI/ANS-58.9-1981 and IEEE 379-1994).
- Assumption (4) of section 1.3 does not apply. The operators will be qualified in accordance with the WTP training program, per Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), section 7.2.
- "Automatic reactor trip…" does not apply.

5.4 ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems

Terminology

- "Containment" or "containment vessel" is interpreted to mean "confinement".
- "Seismic Category I standards" is interpreted as seismic requirements for a SSC with a seismic safety function per SRD Volume II (24590-WTP-SRD-ESH-01-001-02) Safety Criterion 4.1-3 for the WTP.
- "Safety related" for the purposes of this standard is interpreted to mean "SDC or SDS", as appropriate.
- "Technical specification(s)" is interpreted to mean "Technical Safety Requirements" or "TSR(s)".
- "Condition I" is interpreted for WTP to mean "normal operation".
- "Safety-related function" for the purposes of implementation of this standard is interpreted to mean "safety function" as defined in DOE/RL-96-0006, Rev 2, performed by SDC, SDS, safety-class or safety-significant SSCs.
- In definition of "single failure", reference [1] does not apply to WTP.
- Safety classes 1, 2, and 3 (section 4.5) are interpreted to be SDC, SDS safety-class or safety-significant systems.

Non-Applicability

- For WTP, the need for emergency onsite power will be ascertained in accordance with the DOE/RL-96-0004 process as part of determining hazard control strategies.

- In the definition of "short term" (section 2), everything after "...up to 24 hours following an initiating event" applies to nuclear power reactor plants and is therefore not applicable to WTP.
- Sections 3.1 through 3.3 of ANSI/ANS 58.9 are not applicable to the WTP. Applicability of the single failure criteria to the work and hazards presented by the WTP is described in section 3.0 of this Implementing Standard.
- Reactor-specific regulations (e.g., 10 CFR 50 Appendix A) are not applicable to WTP (see section 1, 1st paragraph).
- References to a reactor "unit", "safe shutdown", and "loss of coolant accident" are nuclear reactor plant-specific and, therefore, do not apply to WTP.
- Sections 3.1 through 3.3 are reactor-specific and do not apply to WTP.

5.5 IEEE STD 379-1994, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems (This section has been deleted)

Refer to Section 24 of Appendix C for the tailoring of IEEE STD 379-1994.

5.6 IEEE STD 1023-1988, IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations

Terminology

- "Nuclear power generating stations" is interpreted to mean a nuclear facility such as WTP.

Non-Applicability

- Application of the formal human factors engineering process described in subsection 6.1.1 of IEEE Std 1023-1988 is tailored to the work and hazards presented by the WTP in subsection 2.6.2 of this Implementing Standard.

5.7 ISA-S84.01-1996, Safety Instrumented Systems for the Process Industries

Terminology

- The definition of "common-cause failure" given in DOE/RL-96-0006 is used, rather than that in section 3 of the consensus standard.
- "Safety Instrumented System (SIS)" for the purpose of this standard is interpreted to mean any instrumentation and control system that is SDC or SDS, as appropriate.

6.0 References

Project Documents

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan

Code and Standards

10 CFR 50, Domestic Licensing of Production and Utilization Facilities, *Code of Federal Regulations*, as amended.

10 CFR 830, Subpart A, Quality Assurance Requirements, *Code of Federal Regulations*, as amended.

10 CFR 835, Occupational and Radiological Protection, *Code of Federal Regulations*, as amended.

AICHE, 1992, American Institute of Chemical Engineers (AIChE), *Guidelines for Hazard Evaluation Procedures*, Second Edition with Worked Examples, Cnter for Chemical process Safety, New York, New York, USA 1992.

ANSI/ANS-51.1-1983, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*. American Nuclear Society, La Grange Park, Illinois.

ANSI/ANS-52.1-1983, *Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants*. American Nuclear Society, La Grange Park, Illinois.

ANSI/ANS-58.8-1994, *Time Response Desing Criteria for Safety-Related Operator Actions*. American Nuclear Society, La Grange Park, Illinois.

ANSI/ANS-58.9-1981, *Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems*. American Nuclear Society, La Grange Park, Illinois.

DOE G 225.1A-1, *Guide for DOE O 225.1A, Accident Investigations*, US Department of Energy, Washington D.C.

DOE G 420.1-1, *Nonreactor Nuclear Safety Criteria and Explosives Safety Criteria Guide for use with DOE O 420.1, Facility Safety*, US Department of Energy, Washington D.C.

DOE G 420.1-1, *Facility Safety*, Chg 2. US Department of Energy, Washington D.C.

DOE O 6430.1A, *General Design Criteria*, US Department of Energy, Washington D.C.

DOE/RL-96-0004, Revision 2, *Process for Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for the RPP Waste Treatment Plant Contractor*, US Department of Energy, Richland, Washington.

DOE/RL-96-0006, Revision 2, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for the RPP Waste Treatment Plant Contractor*, US Department of Energy, Washington, D.C.

IEEE Std 1023-1988, *IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations*, Piscataway, New Jersey.

IEEE Std 379-1994, *IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Stations Safety Systems*, Piscataway, New Jersey.

IEEE Std 384-1992, *IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits. IEEE Power Engineering Society*, Piscataway, New Jersey.

ISA-S84.01-1996, *Application of Safety Instrumented Systems for the Process Industries*. Instrument Society of America, Pittsburgh, Pennsylvania.

Other Documents

Modarres M, 1993. *What Every Engineer Should Know about Reliability and Risk Analysis*, Marcel Dekker Inc., New York, New York.

Appendix C

Implementing Standards

CONTENTS

1.0 ISO 10007:1995(E), Quality Management - Guidelines for Configuration Management

Revision: First Edition, 15 April 1995
Sponsoring Organization: International Organization for Standardization

WTP Specific Tailoring

The following tailoring of ISO 10007:1995(E) is required for use by the WTP contractor as an Implementing Standard for Configuration Management.

Page 1, Section 1 Scope

Delete the last sentence in the second paragraph.

Justification: The WTP Project has not adopted ISO 9001, ISO 9002, ISO 9003, and ISO 9004 as implementing standards.

Page 1, Section 2 Normative References

Delete reference to the ISO 10011 series of standards.

Justification: As discussed for Section 8, for WTP the approved QAM defines the principles, criteria, and practices for the configuration management system audit.

Page 1, Section 3 Definitions

Delete definition 3.4, "configuration board", and Note 2.

Justification: Deletes definition and note dealing with a configuration board to be consistent with deletion of ISO 10007:1995(E) section 7.3, "Configuration board".

Page 4, Section 6.2 Structure of configuration management

Delete '(normally a "configuration board")' in the 2nd to last paragraph.

Justification: Deletes words dealing with a configuration board to be consistent with deletion of ISO 10007:1995(E) section 7.3, "Configuration board".

Page 5, Section 7.3 Configuration board

Delete this section in its entirety.

Justification: The project manager has not established a configuration board as permitted. Equivalent functions and responsibilities of a configuration board exist in the change control processes for WTP design, AB Document Maintenance, and project interfaces, and in the QAM for Plant Installed Software.

Page 7, Section 7.7 Configuration Management Plan (CM Plan)

Delete second paragraph

Justification: This paragraph addresses activities outside the WTP project workscope or control (i.e., multiple projects, multi-level contracts, and customer configuration management plans).

Page 8, Section 8 Configuration Management System Audit

Revise the last paragraph to read:

"Principles, criteria, and practices of the CM system audit should comply with the Quality Assurance Manual."

Justification: For WTP the approved QAM defines the principles, criteria, and practices for the conduct of audits and self-assessments.

Page 9, Annex A, Section A2 Policies and procedures

Delete all words following "the CM organization" in the 2nd subparagraph.

Justification: Deletes words dealing with a configuration board to be consistent with deletion of ISO 10007:1995(E) section 7.3, "Configuration board".

Page 9, Annex A, Section A4 Configuration control

Delete all words after "the organization" in the first subparagraph.

Justification: Deletes words dealing with a configuration board to be consistent with deletion of ISO 10007:1995(E) section 7.3, "Configuration board".

Page 11 and 12, Annex B

Delete.

Justification: Although provided only as information, as noted in Section 1 above, the ISO 9000 Series of Standards are not being implemented at WTP and this Annex is therefore removed to reduce potential confusion to non-applicable cross references.

2.0 DOE-STD-1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities"

Revision: Change Notice #1 dated 1/96 and DOE Newsletter dated 1/22/98 (Interim Advisory on Straight Winds and Tornados)

Sponsoring Organization: DOE

WTP Specific Tailoring

The following tailoring of DOE-STD-1020-94 is required for use by the WTP contractor as an Implementing Standard for seismic analysis and design.

Page 1-6, Section 1.3 Evaluation of Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the WTP is a new facility.

Page 2-1, Section 2.2 General Approach for Seismic Design and Evaluation

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per section 3.7.2 of NRC NUREG-0800, Revision 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Use ASCE 4-98 in lieu of ASCE 4-86.

Justification: ASCE 4-98 is more current.

Page 2-6, Section 2.3 Seismic Design and Evaluation of Structures, Systems, and Components

SDC/SDS/RRC SSCs:

Perform performance categorization of SSCs per SRD Safety Criterion 4.1-3 in lieu of DOE-STD-1021-93.

SC/SS/APC SSCs:

Perform performance categorization of SSCs per Section 2.4 of DOE-STD-1021-93 (reaffirmed April 2002) as described in SRD Safety Criterion 4.1-3.

Justification: For SDC/SDS/RRC SSCs, a more conservative approach than required by DOE-STD-1021-93 is adopted. This approach is implemented by SRD Safety Criterion 4.1-3 and Appendix A to Volume II of the SRD. For SC/SS/APC SSCs, DOE-STD-1021-93 is directly linked to DOE Order 420.1A, which is invoked by 10 CFR 830 as the source of nuclear safety design criteria.

Page 2-8, Section 2.3.1 Performance Category 1 and 2 Structures, Systems, and Components

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page 2-12, Section 2.3.2 Performance Category 3 and 4 Structures, Systems, and Components

Disregard the requirements for PC-4 SSCs.

Justification: There are no PC-4 SSCs at the WTP.

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per section 3.7.2 of NRC NUREG-0800, Revision 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Use ACI 349 for design of reinforced concrete in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Use ANSI/AISC N690 for design of structural steel in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Page 2-15, Section 2.3.3 Damping Values for Performance Category 3 and 4 Structures, Systems, and Components

Use ASME Code Case N-411 damping value for piping in lieu of those shown in Table 2-3.

Justification: This value is acceptable to the NRC for nuclear power plants.

Page 2-18, Section 2.4.1 Equipment and Distribution Systems

Perform seismic design of PC-1 and -2 elements of structures and equipment per the provisions of 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page 2-22, Section 2.4.2 Evaluation of Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the WTP is a new facility.

Page 2-24, Section 2.5 Summary of Seismic Provisions

Disregard the requirements for PC-4 SSCs.

Justification: There are no PC-4 SSCs at the WTP.

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per section 3.7.2 of NRC NUREG-0800, Revision 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Use the seismic provisions in Table 2-5 concerning PC-3 SSCs except that the structural capacity is to be based on code ultimate strength or allowable behavior level.

Justification: Limit-state level method of determining the structural capacity is more appropriate for evaluation of existing facilities (the WTP is a new facility).

Page 3-1, Section 3.1 Introduction

SDC/SDS/RRC SSCs:

Perform performance categorization of SSCs per SRD Safety Criterion 4.1-3 in lieu of DOE-STD-1021-93.

SC/SS/APC SSCs:

Perform performance categorization of SSCs per Section 2.4 of DOE-STD-1021-93 (reaffirmed April 2002) as described in SRD Safety Criterion 4.1-3.

Justification: For SDC/SDS/RRC SSCs, a more conservative approach than required by DOE-STD-1021-93 is adopted. This approach is implemented by SRD Safety Criterion 4.1-3 and

Appendix A to Volume II of the SRD. For SC/SS/APC SSCs, DOE-STD-1021-93 is directly linked to DOE Order 420.1A, which is invoked by 10 CFR 830 as the source of nuclear safety design criteria.

Page 3-2, Section 3.2 Wind Design Criteria

Use peak gust speed values contained in Attachment "A" of DOE Interim Advisory dated 1/22/98 in lieu of fastest-mile wind speeds shown in Table 3-2; also, per DOE Interim Advisory, use an importance factor for PC-2 SSCs of 1.0 in lieu of 1.07 indicated in Table 3-1.

Justification: The Newsletter was issued by DOE as an interim measure for use with DOE-STD-1020-94 until such time as the standard is revised.

Page 3-5, Section 3.2.1 Performance Category 1

Design structural steel PC-1 structures per AISC Manual of Steel Construction, Allowable Stress Design, Ninth edition.

Justification: The AISC code is preferred to the UBC because it is a national consensus code.

Design reinforced concrete PC-1 structures per ACI 318-99.

Justification: The ACI 318 code is preferred to the UBC because it is a national consensus code.

Page 3-6, Section 3.2.2 Performance Category 2

Design structural steel PC-2 structures per AISC Manual of Steel Construction, Allowable Stress Design, Ninth edition.

Justification: The AISC code is preferred to the UBC because it is a national consensus code.

Design reinforced concrete PC-2 structures per ACI 318-99.

Justification: The ACI 318 code is preferred to the UBC because it is a national consensus code.

Page 3-6, Section 3.2.3 Performance Category 3

Design structural steel PC-3 structures per ANSI/AISC N690-94.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Design reinforced concrete PC-3 structures per ACI 349-97.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Disregard requirements for tornado design.

Justification: Tornado is not a credible NPH at the WTP site.

Page 3-11, Section 3.2.4 Performance Category 4

Delete this section.

Justification: There are no PC-4 SSCs at the WTP.

Page 3-13, Section 3.3 Evaluation of Existing SSCs

Delete this section.

Justification: This section deals with existing facilities and the WTP is a new facility.

Page 4-1, Section 4.0 Flood Design and Evaluation Criteria

Disregard criteria for the design of SSCs for river flooding.

Justification: River flooding is not a credible NPH at the WTP site, and only the criteria dealing with local precipitation that affects roof design and site drainage are applicable to the WTP design.

Page 4-4, Section 4.1.2 Flood Evaluation Process

SDC/SDS/RRC SSCs:

Perform performance categorization of SSCs per SRD Safety Criterion 4.1-3 in lieu of DOE-STD-1021-93.

SC/SS/APC SSCs:

Perform performance categorization of SSCs per Section 2.4 of DOE-STD-1021-93 (reaffirmed April 2002) as described in SRD Safety Criterion 4.1-3.

Justification: For SDC/SDS/RRC SSCs, a more conservative approach than required by DOE-STD-1021-93 is adopted. This approach is implemented by SRD Safety Criterion 4.1-3 and Appendix A to Volume II of the SRD. For SC/SS/APC SSCs, DOE-STD-1021-93 is directly linked to DOE Order 420.1A, which is invoked by 10 CFR 830 as the source of nuclear safety design criteria.

Page 4-12, Section 4.2.4 Performance Category 4

Delete this section.

Justification: There are no PC-4 SSCs at the WTP.

Page 4-13, Section 4.3.3 Site Drainage and Roof Design

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page 4-15, Section 4.4 Considerations for Existing Construction

Delete this section.

Justification: This section deals with existing facilities and the WTP is a new facility.

Page 4-16, Section 4.5 Probabilistic Flood Risk Assessment

Do not perform a probabilistic flood risk assessment of the WTP site.

Justification: UCRL-21069, "Probabilistic Flood Hazard Assessment for the N Reactor, Hanford, Washington", July 1988, contains a probabilistic flood risk assessment of the N reactor site. The WTP site is close to the N Reactor site (about 10 miles away) and further away from the Columbia River. Therefore, the N Reactor flood assessment may be used and no assessment of the WTP site is required.

Page B-4, App. B, Section B.2 Graded Approach, Performance Goals, and Performance Categories

SDC/SDS/RRC SSCs:

Perform performance categorization of SSCs per SRD Safety Criterion 4.1-3 in lieu of DOE-STD-1021-93.

SC/SS/APC SSCs:

Perform performance categorization of SSCs per Section 2.4 of DOE-STD-1021-93 (reaffirmed April 2002) as described in SRD Safety Criterion 4.1-3.

Justification: For SDC/SDS/RRC SSCs, a more conservative approach than required by DOE-STD-1021-93 is adopted. This approach is implemented by SRD Safety Criterion 4.1-3 and Appendix A to Volume II of the SRD. For SC/SS/APC SSCs, DOE-STD-1021-93 is directly linked to DOE Order 420.1A, which is invoked by 10 CFR 830 as the source of nuclear safety design criteria.

Page B-8, App. B, Section B.3 Evaluation of Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the WTP is a new facility.

Page C-1, App. C, Section C.1 Introduction

SDC/SDS/RRC SSCs:

Perform performance categorization of SSCs per SRD Safety Criterion 4.1-3 in lieu of DOE-STD-1021-93.

SC/SS/APC SSCs:

Perform performance categorization of SSCs per Section 2.4 of DOE-STD-1021-93 (reaffirmed April 2002) as described in SRD Safety Criterion 4.1-3.

Justification: For SDC/SDS/RRC SSCs, a more conservative approach than required by DOE-STD-1021-93 is adopted. This approach is implemented by SRD Safety Criterion 4.1-3 and Appendix A to Volume II of the SRD. For SC/SS/APC SSCs, DOE-STD-1021-93 is directly linked to DOE Order 420.1A, which is invoked by 10 CFR 830 as the source of nuclear safety design criteria.

Page C-19, App. C, Section C.3.2 Earthquake Ground Motion Response Spectra

Disregard section C.3.2.1 discussion and Table C-4. Follow 1997 UBC for the WTP design.

Justification: Section C.3.2.1 discussion and Table C-4 are based on 1994 UBC; the 1997 UBC is more current.

Page C-27, App. C, Section C.4 Evaluation of Seismic Demand (Response)

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page C-29, App. C, Section C.4.1 Dynamic Seismic Analysis

Use ASCE 4-98 in lieu of ASCE 4-86.

Justification: ASCE 4-98 is more current.

Page C-31, App. C, Section C.4.2 Static Force Method of Seismic Analysis

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page C-32, App. C, Section C.4.3 Soil-Structure Interaction

Use ASCE 4-98 in lieu of ASCE 4-86.

Justification: ASCE 4-98 is more current.

Page C-38, App. C, Section C.4.4 Analytical Treatment of Energy Dissipation and Absorption

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per section 3.7.2 of NRC NUREG-0800, Revision 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Page C-52, App. C, Section C.5.1 Capacity Approach

Use ACI 349 for design of reinforced concrete in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Use ANSI/AISC N690 for design of structural steel in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Page C-62, App. C, Section C.7 Special Considerations for Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the WTP is a new facility.

Page C-66, App. C, Section C.9 Alternate Seismic Mitigation Measures

Delete this section.

Justification: Seismic base isolation is not planned to be used in the WTP design.

Page D-3, App. D, Section D.3 Load Combinations

Design structural steel PC-1 and PC-2 structures per AISC Manual of Steel Construction, Allowable Stress Design, Ninth edition.

Justification: The AISC code is preferred because it is a national consensus code.

Design reinforced concrete PC-1 and PC-2 structures per ACI 318-99.

Justification: The ACI 318 code is preferred because it is a national consensus code.

Design structural steel PC-3 SSCs structures per ANSI/AISC N690-94.

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

Design reinforced concrete PC-3 SSCs structures per ACI 349-97

Justification: This change is made for consistency with NRC acceptance criteria contained in section 3.8.4 of NUREG-0800, Revision 2 (Draft).

3.0 ANSI/AISC N690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities"

Revision: 1994
Sponsoring Organization: American National Standards Institute/American Institute of Steel Construction

WTP Specific Tailoring

The following tailoring of ANSI/AISC N690 is required for use by the WTP contractor as an Implementing Standard for structural design.

Page 22, Section Q1.5.7.1 Primary Stresses

Revise the stress limit coefficients for compression in Table Q1.5.7.1 as follows:

- 1.3 instead of 1.5 [stated in footnote (c)] in load combinations 2, 5, and 6
- 1.4 instead of 1.6 in load combinations 7, 8, and 9
- 1.6 instead of 1.7 in load combination 11

Justification: These changes are made for consistency with the NRC requirements of Appendix F of section 3.8.4 of NUREG-0800 (Draft Rev. 2).

Page 22, Section Q1.5.7.1 Primary Stresses

Delete the following load combinations:

4. D + L + Eo
6. D + L + Ro + To + Eo

Justification: These load combinations are requried for evaluation of an Operation Basis Earthquake (OBE). The WTP project has not identified an OBE event.

This section has been deleted.

This section has been deleted.

6.0 NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Materials

Revision: 2003 edition

Sponsoring Organization: National Fire Protection Association

WTP Specific Tailoring

The following tailoring of NFPA 801-03 is required for use by the WTP project as an implementing standard for fire safety.

Section 4.3, Fire Protection Program

NFPA 241 does not apply to the WTP Project until Hot Commissioning.

Justification: The WTP Project is following 29 CFR 1926, Subparts F and J as referenced in the Non-radiological Worker Health and Safety Plan.

Section 5.5

Replace NFPA 220 with IBC 2000.

Justification: The applicable building code for the WTP Project is the 1997 Uniform Building Code (UBC). UBC specifies building requirements for fire resistance, allowable floor area, building height limitations, and building separation.

Section 5.13.2

Change the code edition of NFPA 70 from 2002 to 1999.

Justification: NFPA 801, in all versions, simply refers to NFPA 70 for electrical installation. It does not make any special concessions to NFPA 70. The 2002 version of NFPA 70 does not emphasize additional critical safety requirements that would adversely impact the safety of the design of a nuclear waste treatment plant. NFPA 70-2002 would, however, put an undue cost and schedule impact onto the Project based on the present state of the design. Since the 1999 version of NFPA 70 has previously been deemed to provide adequate safety and no critical items have been identified in the 2002 edition, the project should continue using the 1999 edition since it is more cost effective.

Section 6.3, 6.8.1 and 6.8.2

Change the code edition of NFPA 72 from 1999 to 2002.

Justification: The WTP Project has chosen to use the 2002 code edition of NFPA 72. Using the 2002 version of NFPA 72 is as or more safe and is more cost effective than the 1999 version of the standard. NFPA 72-02 is inherently more safe than the 1999 version because it has been revised to incorporate requirements for safer and more reliable technology not previously available. Plus, NFPA 72-02 allows the Project more design possibilities (spacing, routing, power supplies, etc.) based on the type of detection systems used. By designing detection systems based more on the system's use the Project can take advantage of cost savings.

Section 8.1.2

NFPA 241 does not apply to the WTP Project until Hot Commissioning.

Justification: The WTP Project is following 29 CFR 1926, Subparts F and J as referenced in the Non-radiological Worker Health and Safety Plan.

Appendix A, Section 3.3.23, Noncombustible

Replace NFPA 220 with IBC 2000.

Justification: The applicable building code for the WTP Project is the 1997 Uniform Building Code (UBC). UBC specifies building requirements for fire resistance, allowable floor area, building height limitations, and building separation.

Appendix C, Section 8.2

Replace NFPA 220 with IBC 2000.

Justification: The applicable building code for the WTP Project is the 1997 Uniform Building Code (UBC). UBC specifies building requirements for fire resistance, allowable floor area, building height limitations, and building separation.

Appendix D, Section 1.1, NFPA Publications

Replace NFPA 220 with IBC 2000.

Justification: The applicable building code for the WTP Project is the 1997 Uniform Building Code (UBC). UBC specifies building requirements for fire resistance, allowable floor area, building height limitations, and building separation.

7.0 ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures

Revision: 2001
Sponsoring Organization: American Concrete Institute

WTP Specific Tailoring

The following tailoring of ACI 349-01 is required for use by the WTP contractor as an implementing standard for structural design.

Chapter 21

Replace Chapter 21 of ACI 349-01 with Chapter 21 of ACI 318-99, while maintaining the following specific provisions of ACI 349-01 Chapter 21 as identified in:

- Section 21.2.7 (anchorage)
- Section 21.6.1 (height/length criteria)

Justification: Chapter 21 of ACI 349-01 is based on criteria from ACI 318-95. The American Concrete Institute completed a major revision of ACI 318 between the years 1995 and 1999 with respect to seismic proportioning and detailing. The WTP project wishes to adopt the most current methodology for seismic detailing as presented in ACI 318-99 Chapter 21 pertaining to structures in high seismic risk region, in lieu of that presented in ACI 349-01 Chapter 21.

The HLW and Pretreatment reinforced concrete structures (designated Seismic Category I) of the WTP project are large shear wall and slab structures of heavy proportions, which exhibit small lateral deflections. ACI 349-01 Chapter 21 describes that at a height-to-length (h/l) ratio of less than 2, the concrete walls act in shear with insignificant bending deformation, thus boundary elements are not required. This criteria, along with the requirements for anchorage are key elements of the ACI 349-01 design philosophy contained in Chapter 21.

The purpose of maintaining the specific sections of ACI 349-01 Chapter 21 as cited above is to ensure that the specific provisions of ACI 349-01 are maintained while incorporating the more current methodology for seismic detailing requirements of ACI 318-99.

Notes:

1. For the purpose of determining the need for boundary elements, the $h_{w/}l_w$ criterion of ACI 349-01 shall be applied for the entire wall (where h_w shall be defined as the total height of the wall and l_w shall be defined as the length of the wall).
2. For the purpose of determining the need for boundary elements using the $0.2f'_c$ criterion, the compressive stress in the shear wall (or shear wall segment) shall be determined by considering the axial compression and in-plane bending behavior of the wall (or shear wall segment) acting as a

2. For the purpose of determining the need for boundary elements using the $0.2f'_c$ criterion, the compressive stress in the shear wall (or shear wall segment) shall be determined by considering the axial compression and in-plane bending behavior of the wall (or shear wall segment) acting as a "beam". The maximum compressive stress may be determined by using the formula, $P/A \pm MC/I$ (where C is lever arm or the distance from neutral axis to the extreme fiber, A is the area of column, and I is the second moment of area) based on the axial loads (i.e., P) and moments (i.e., M) computed by integrating the stresses obtained from an explicit finite element model (e.g., GTSTRUDL model) and assuming a rectangular cross section for the shear wall (or shear wall segment). Alternatively, the "beam" properties may include the effects of the cross walls, in which case the axial loads (i.e., P) and moments (i.e., M) shall be computed by including the stresses on the cross walls.

3. Strain Criteria (Tailoring of ACI 318-99)

Chapter 21 Section 21.6.6.3 (Walls)

In addition to the provisions of this section, boundary elements are not required when the concrete compressive strain, resulting from the worst case loading combination, does not exceed 0.002.

Justification: Continued use of a concrete compressive stress limit of $0.2f_c'$ for wall boundary element requirements has been determined to be very conservative. Therefore, a special system of design that utilizes a concrete compressive strain limit of 0.002 for wall boundary element requirements is warranted. For further discussion, see 24590-HLW-RPT-CSA-03-013.

Chapter 21 Section 21.7.5.3 (Diaphragms)

In addition to the provisions of this section, boundary elements are not required when the concrete compressive strain, resulting from the worst case loading combination, does not exceed 0.002.

Justification: Continued use of a concrete compressive stress limit of $0.2f_c'$ for diaphragm boundary element requirements has been determined to be very conservative. Therefore, a special system of design that utilizes a concrete compressive strain limit of 0.002 for diaphragm boundary element requirements is warranted. For further discussion, see 24590-HLW-RPT-CSA-03-014.

Chapter 21 Section 21.7.8.1 (Diaphragms)

In lieu of the provisions of this section, proportion reinforcement across the entire width of the diaphragm to resist the factored axial forces and moments acting in the plane of the diaphragm.

Justification: The finite element analysis is the best available description of the structural response of the slabs acting as diaphragms to the various load combinations. Therefore, the finite element results will be utilized to determine the stress distribution across the entire width of the diaphragm. Placement of reinforcement will be distributed accordingly.

Note: Item 3 is not applicable to BOF facilities due to the simplicity of the analysis and design of BOF facilities.

8.0 ACI 318, Building Code Requirements for Structural Concrete and Commentary

Revision: 1999
Sponsoring Organization: American Concrete Institute

<u>WTP Specific Tailoring</u>

The following tailoring of ACI 318-99 is required for use by the WTP contractor as an implementing standard for design of reinforced concrete for Seismic Category III SSCs, as noted.

Chapter 9, Section 9.2 Required Strength

The following additional load combinations from the *Uniform Building Code*, 1997, section 1612.2.1, shall be included in the load combinations evaluated for design of reinforced concrete:

Equation (12-5): $1.2D + 1.0E + (f_1L + f_2S)$

Equation (12-6): $0.9D \, \forall \, (1.0E \text{ or } 1.3W)$

Justification: The additional load combinations implemented are not identified in the ACI load combinations. These combinations are evaluated to ensure adequate equivalency with commercial design in accordance with the UBC.

Chapter 21, Section 21.2.1.3

Seismic detailing requirements for "moderate seismic risk" will be used.

Justification: The "moderate seismic risk" classification is consistent with the Seismic Category III, which is an important facility in seismic zone 2B.

General (no specific chapter)

Design of concrete anchorage will follow the requirements of ACI 349-01, Appendix B.

Justification: This design standard represents the current industry approach to design of concrete embedments. This design method has been adopted by ACI 349 committee and used in the 2001 edition for Appendix B. The load factors are lower than those identified for safety related structures applicable to higher seismic classification. The load factors in this publication are appropriate for use in important commercial structures commensurate with SC-III.

9.0 AISC M016, Manual of Steel Construction, Allowable Stress Design (ASD)

Revision: 9th Edition
Sponsoring Organization: American Institute of Steel Construction

WTP Specific Tailoring

The following tailoring of M016 is required for use by the WTP contractor as an implementing standard for design of structural steel for Seismic Category III SSCs.

No specific section

Load combinations for design of structural steel members utilize those identified in UBC 97, section 1612.3.

Justification: These load combinations represent the commercial requirements for allowable stress design of structural steel. Use of these load combinations will ensure compliance with the commercial design in accordance with the UBC.

No specific section

Seismic detailing requirements shall be in accordance with UBC 97, Chapter 22, Division V, section 2214, for moderate seismic risk structures.

Justification: The requirements contained in this section contain accepted industry practice for design of important commercial steel structures. Use of this section will ensure compliance with the commercial design in accordance with the UBC.

10.0 UBC 97, Uniform Building Code

Revision: 1997
Sponsoring Organization: International Conference of Building Officials

WTP Specific Tailoring

The following tailoring of UBC 97 is required for use by the WTP contractor as an implementing standard for design of reinforced concrete for Seismic Category III SSCs, as noted.

Division II Snow

Design for snow loads shall be in accordance with ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, section 7.0, utilizing ground snow loads identified in Safety Criterion 4.1-3.

Justification: This approach to design of snow loads is an acceptable industry practice for evaluation of structures under snow loads. This code is more thorough in its consideration of these loads than the UBC methodology.

Division III Wind

Design for wind loads shall be in accordance with ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, section 6.0, utilizing 3-second gust values identified in Safety Criterion 4.1-3.

Justification: This approach to design of wind loads is an acceptable industry practice for evaluation of structures under wind loads. This code is more thorough in its consideration of these loads than the UBC methodology.

The following tailoring of UBC 97 is required for use by the WTP contractor as a daughter standard referenced by the implementing standard for the fire protection, as noted.

Chapters 1 through 15 and 24 through 35

Applicable to the process buildings (LAW, HLW, and PT) and the Analytical Laboratory Facility, replace Chapters 1 through 15 and 24 through 35 of the 1997 UBC with corresponding Chapters of the 2000 International Building Code (IBC).

Justification: For the process buildings (LAW, HLW, and PT) and the Analytical Laboratory Facility, the non-structural portions of the 1997 UBC are updated to the 2000 IBC. The 2000 IBC is the follow on model building code to 1997 UBC and replaces the UBC.

This section has been deleted

12.0 IEEE-387, Standard Criteria For Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations

Revision: 1995
Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

WTP Specific Tailoring

The following tailoring of IEEE-387 is required for use by the WTP project as an implementing standard for the SDC electrical power system design.

Pages 1 – 40, All Sections   Clarification of Nuclear Power Generating Station Terminology

The terms "Standby Power", "Standby Power Supply" in the Standard apply to the "Emergency Power" or "Emergency Power Supply" in the WTP.

Justification: As determined by the ISM review process, the Standby Generators on the RPP-WTP are not classified as SDC while the Emergency Generators are classified as SDC.

The terms "nuclear plant", "nuclear power generating stations" and "conventional plant" will be taken to mean the WTP.

Justification: Clarifies how the standard will apply to the WTP project.

Page 3, Section 1.1.3(c)   Exclusions

Remove "day tank" as an exclusion.

Justification: The day tank is listed in section 1.1.1 as an inclusion. This change clarifies the scope of the standard.

Page 4, Section 1.2   Purpose

Replace the last words "the design basis events cataloged in the Plant Safety Analysis." with "the design basis events as determined by the ISM review process".

Justification: For the WTP project, the design basis is determined during the ISM review and is not cataloged in a plant safety analysis.

Page 4, Section 2 References

Delete IEEE Std 603-1991, *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations*.

Justification: This standard will be replaced by ANSI/ISA-S84.01-1996, Per 24590-WTP-ABCN-ESH-01-027.

The following reference standard shall be included:

ANSI/ISA-S84.01-1996, *Application of Safety Instrument Systems for Process Industries*.

Justification: ANSI/ISA-S84.01-1996 replaces IEEE-603 for the WTP, per 24590-WTP-ABCN-ESH-01-027.

DOE/RL-96-0006, Revision 1, *Top-level radiological, nuclear, and process safety standards and principles for TWRS privatization contractors*.

Justification: This is a regulatory basis document for the WTP per the SRD.

Page 5, Section 3 Definitions

3.3 Design Basis Events (DBE): replace definition in the standard with the following: "Postulated events providing bounding conditions for establishing the performance requirements of structures, systems, and components that are necessary to: 1) ensure the integrity of the safety boundaries protecting the worker; 2) place and maintain the facility in a safe state indefinitely; or 3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design-Basis Events also establish the performance requirements of the structures, systems and components whose failure under Design-Basis Event conditions could adversely affect any of the above functions."

Justification: This definition is from DOE/RL-96-0006.

3.4 Design Load: Replace the words "during and following shutdown of the reactor", from the definition and replace with "during a DBE".

Justification: The reference to shutdown of the reactor is applicable to a nuclear power generating facility and is not applicable to the WTP.

The following definitions shall be modified to read as follows.

3.12 Standby Power Supply: This definition applies to the Emergency Power Supply for the WTP.

Justification: The WTP has a Standby Power Supply which is not SDC. This standard shall be applied to the Emergency Power Supply on the WTP which is SDC.

The following term shall be added.

3.15 Emergency Power Supply: The power supply that is selected to furnish electrical energy to the SDC power distribution system when the offsite power source is not available.

Justification: As determined by the ISM review process, the Standby Power Supply on the WTP is not classified as SDC while the Emergency Power Supply is classified as SDC.

Pages 7 - 9, Section 4.4, Table 1 Design and Application Considerations

For Item 46, replace with the following: "Monitoring diesel-generator units during a design basis event."

For Item 49, replace with the following: "Communication means between the diesel-generator enclosure and the main control room."

Justification: Item 46 refers to accident and post accident conditions which are not clearly defined for the WTP. The term "design basis events" is more applicable to the WTP. Item 49 refers to a diesel-generator room. The emergency diesel-generators for the WTP will be within pre-fabricated, weather-proof enclosures. Therefore, this term is not applicable to the WTP.

Page 10, Section 4.5.2.3 Control Points

Replace with the following: "The emergency diesel generator will be automated and indication of the safety functions shall be provided to the main control room. Manual control and indication shall be provided external to the main control room."

Justification: The control philosophy for a Nuclear Power Generating Station is not applicable for the WTP project. A Nuclear Power Generating Station has one main control room, with hard wired controls for all major equipment. The WTP project has separate control rooms for each facility and automated controls to minimize human factor errors.

Page 11, Section 4.5.4 Protection

Replace the terms "accident conditions" and "non-accident conditions" with "design basis event" and "non design basis event".

Justification: The terms "accident conditions" and "non-accident conditions" are not clearly defined for the WTP. The term "design basis events" is more applicable to the WTP.

Page 24, Section 7.5.5 Safety Injection Actuation Signal (SIAS) Test

Not applicable for the WTP.

Justification: There is no safety injection actuation signal for WTP. This section is specific to the actuation of safety injection systems which require power for the operation of safety injection equipment in Nuclear Power Generating Stations.

Page 24, Section 7.5.6 Combined SIAS and LOOP Test

Not applicable for the WTP.

Justification: There is no SIAS (safety injection actuation signal) for WTP. This section is specific to the actuation of safety injection systems in Nuclear Power Generating Stations.

13.0 IEEE-741, Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations

Revision: 1990
Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

WTP Specific Tailoring

The following tailoring of IEEE-741 is required for use by the WTP project as an implementing standard for the SDC electrical power system design.

Page 1, Section 2 References

The following references shall be excluded:

IEEE Std-317-1983 (Reaff 1996), *IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations, (ANSI).*

Justification: Containment penetration assemblies are unique to the containment structure of Nuclear Power Generating Stations and have no equivalent in the WTP project.

IEEE Std 415-1986, (Reaff 1993), *IEEE Guide for Planning of Pre-Operational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Stations.*

Justification: This standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for pre-operational testing programs will be developed internally for the WTP project.

IEEE Std 765-1995, *IEEE Standard for Preferred Power Supply for Nuclear Power Generating Stations.*

Justification: This standard addresses the normal offsite power for a nuclear power generating facility. The design of the offsite power distribution system has been coordinated with the DOE, (ref. ICD-11).

The following reference standard shall be included:

ANSI/ISA-S84.01-1996, *Application of Safety Instrument Systems for Process Industries.*

Justification: ANSI/ISA-S84.01-1996 replaces IEEE-603 for the WTP, per 24590-WTP-ABCN-ESH-01-027.

Page 3, Section 3.0 Definitions

The following terms shall be added.

3.5 Execute Features: The electrical and mechanical equipment and interconnection that perform a function, associated directly or indirectly with a safety function, upon receipt of a signal from the sense and command features. The scope of the execute features extends from the sense and command features output to and including the actuated equipment-to-process coupling.

Justification: The standard references IEEE-603 for the definition of this term. The definition of this term, as it is listed in IEEE-603, is included for clarification of how the term applies to this standard. IEEE-603 is not being utilized for the WTP. Standard ANSI/ISA-S84.01-1996 will be used instead for safety system criteria.

3.6 Sense and Command Features: The electrical and mechanical components and interconnections involved in generating those signals associated directly or indirectly with the safety functions. The scope of the sense and command features extends from the measured process variables to the execute feature input terminals.

Justification: The standard references IEEE-603 for the definition of this term. The definition of this term, as it is listed in IEEE-603, is included for clarification of how the term applies to this standard. IEEE-603 is not being utilized for the WTP. Standard ANSI/ISA-S84.01-1996 will be used instead for safety system criteria.

Page 3, Section 4.0 General design criteria

Delete the reference to IEEE 603.

Justification: ANSI/ISA-S84.01-1996 replaces IEEE-603 for the WTP, per 24590-WTP-ABCN-ESH-01-027. The definitions for "sense and command features" and "execute features" are added as part of this tailored standard.

Page 4, Section 5.1.2 Bus voltage monitoring schemes

Replace the first sentence of sub-section (b) with the following: "Upon sensing the preferred power supply degradation, the condition shall be alarmed via the WTP Programmable Protection System (PPS)."

Justification: The control philosophy for a Nuclear Power Generating Station is not applicable for the WTP project. A Nuclear Power Generating Station has one main control room, with hard wired controls for all major equipment. The WTP project has separate control rooms for each facility and automated controls to minimize human factor errors.

Page 5, Section 5.1.4 Standby Power Supply Protection

The term "standby power supply" refers to the emergency diesel generators for the WTP.

Justification: As determined by the ISM review process, the Standby Generators on the WTP are not classified as SDC while the Emergency Generators are classified as SDC.

Page 7, Section 5.4 Primary Containment Electrical Penetration Assemblies

Not applicable for the WTP.

Justification: Containment penetration assemblies are unique to the containment structure of Nuclear Power Generating Stations and have no equivalent in the WTP project.

Page 8, Section 6.2 Preoperational tests

Delete reference to IEEE Std 415.

Justification: This standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for pre-operational testing programs will be developed internally for the WTP project.

14.0 DOE/RL-94-02, Hanford Emergency Management Plan

Revision: 1 February 2001
Sponsoring Organization: DOE

WTP Specific Tailoring

The following tailoring of DOE/RL-94-02, *Hanford Emergency Management Plan*, is required for use by the WTP contractor as an Implementing Standard for the preparation of Reporting and Incident Investigation.

Page 1, Section 1.1 Purpose

In the 1st sentence of the 3rd paragraph change "the provisions of DOE O 151.1" to "emergency management".

Justification: DOE O 151.1 is not a standard imposed on the WTP or committed to in an authorization basis document.

Page 1, Section 1.1 Purpose

In the 2nd sentence of the 3rd paragraph delete "along with DOE Order".

Justification: No definition of other DOE Orders, which may not apply to WTP.

Page 7, Section 1.3.3 Hazards Survey and Hazards Assessment

Delete the 2nd sentence of the 1st paragraph.

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 1, Section 2.0 Emergency Response Organization (Internal)

In the 2nd sentence delete "DOE O 151.1 and other"

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 1, Section 5.1 Notifications

In the 1st sentence of the 2nd paragraph delete "DOE O 151.1 and".

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 1, Section 5.1 Notifications

In the last sentence of the 3rd paragraph delete "DOE O 151.1 and".

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 1, Section 6.1.2 Water/Groundwater Monitoring

In the 1st sentence delete "required by DOE Order 5400.1 (DOE 1990)".

Justification: DOE O 5400.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 7, Section 7.2.2.3 Emergency Response Planning Guidelines 3 (ERPG-3)

In the 1st sentence of the 2nd paragraph delete "DOE 151.1".

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 1, Section 8.1 Emergency Medical Responsibilities

In the 1st sentence of the 1st paragraph delete "in accordance with DOE O 440.1A (or replacement directive)".

Justification: DOE O 440.1A is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 2, Section 9.2 Recovery Planning

In the 1st sentence of the 3rd paragraph delete "in accordance with DOE O 225.1A, *Accident Investigation* (DOE 1997)".

Justification: DOE O 225.1A is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 2, Section 9.2 Recovery Planning

In the 2nd sentence of the 3rd paragraph delete " in accordance with DOE requirements (e.g., DOE O 225.1A and DOE 5480.19) and RLIP 5484.1A, *Environmental Protection, Safety, and Health Protection Information Reporting Requirements* (DOE/RLIP 1981)".

Justification: DOE O 225.1, DOE O 5480.19 and RLIP 5484.1A are not standards imposed on the RPP-WTP or committed to in an authorization basis document.

Page 7, Section 12.10.1.1 Operational Drill

In the last sentence delete "for compliance with DOE 5480.20A, Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities".

Justification: DOE O 5480.20A is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 2, Section 14.1.1 Emergency Management Functions at the US Department of Energy, Richland Operations Office

In the 9th bulleted item delete "DOE O 151.1 and other".

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 3, Section 14.1.1 Emergency Management Functions at the US Department of Energy, Richland Operations Office

In the last paragraph delete "DOE O 151.1 and other".

Justification: DOE O 151.1 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

Page 1, Section 15 References

Delete DOE O 151.1, DOE O 225.1A, DOE O 440.1, DOE O 5400.1, and DOE/RLIP 5484.1A.

Justification: DOE Orders listed are not standards imposed on the RPP-WTP or committed to in an authorization basis document.

Page 6, Appendix A Item for WAC 173-303-340(5)

Delete "If authorities decline, the documentation will be maintained in accordance with DOE/RL-91-28" in the last column.

Justification: DOE/RL-91-28 is not a standard imposed on the RPP-WTP or committed to in an authorization basis document.

15.0 DOE Order 5480.19, Conduct of Operations Requirements for DOE Facilities

Revision: 23 October 2001
Sponsoring Organization: Department of Energy, Office of Nuclear Safety

WTP Specific Tailoring

The following tailoring of DOE Order 5480.19, *Conduct of Operations Requirements for DOE Facilities*, is required for use by the RPP-WTP contractor as an Implementing Standard for Conduct of Operations.

Page 1, Section 3 Definitions

Delete section 3 in its entirety.

Justification: The definitions provided here do not apply to the WTP.

Page 5, Section 5 Requirements

Change Section 5.a to "The contractor shall use this Order and Attachment 1 in the review and development of existing and proposed directives, plans, or procedures relating to the conduct of operations at DOE facilities."

Justification: Clarification change for applicability to the WTP.

Page 5, Section 6 Responsibilities and Authorities

Delete section 6 in its entirety.

Justification: Deleted as not applicable to the WTP and to avoid confusion.

Page 1-12 General Introduction

In the 3rd paragraph delete sentence "It is recognized that these guidelines cross into areas covered by multiple DOE Orders (e.g., DOE O 5480.4 or DOE O 5500)."

Justification: The requirements imposed on the WTP project are provided in the *Safety Requirements Document Volume II* safety criteria.

Page 1-16, Chapter I, Section C Guidelines

In Section C.1, change the last sentence to "Physical security should be in accordance with the WTP Safety and Security Program Plan."

Justification: Changed to reflect WTP contract requirements.

Page 1-22, Chapter II, Section C Personnel Protection

In the 1st paragraph of section C.5, change "5480.11" to "10 CFR 835".

Justification: Changed to reflect WTP contract requirements. DOE O 5480.11 is not a standard imposed on the WTP or committed to in an authorization basis document.

Page 1-37, Chapter VI, Section A Introduction

Replace "DOE O 5000.3A, OCCURRENCE REPORTING AND PROCESSING OF OPERATIONS INFORMATION OF 5/30/90" with "DOE M 232.1-1A, OCCURRENCE REPORTING AND PROCESSING OF OPERATIONS INFORMATION" in the 1st sentence.

Justification: DOE O 5000.3A has been superceded by DOE O 232.1A. DOE M 232.1-1A is the standard imposed on the RPP-WTP and committed to in the authorization basis documents for occurrence reporting.

Page 1-38, Chapter VI, Section C.1 Events Requiring Investigation

Replace "DOE O 5000.3A" with "DOE M 232.1-1A" in the 1st sentence.

Justification: DOE O 5000.3A has been superceded by DOE O 232.1A. DOE M 232.1-1A is the standard imposed on the RPP-WTP and committed to in the authorization basis documents for occurrence reporting.

Page 1-45, Chapter VII, Section A Introduction

Replace "DOE O 5000.3A" with "DOE M 232.1-1A".

Justification: DOE O 5000.3A has been superceded by DOE O 232.1A. DOE M 232.1-1A is the standard imposed on the RPP-WTP and committed to in the authorization basis documents for occurrence reporting.

Page 1-87, Chapter XVI, Section B **Discussion**

In the 1st paragraph, second sentence delete "in accordance with NUREG-0899".

Justification: NUREG-0899 is not a standard imposed on the WTP or committed to in an authorization basis document. The WTP project will provide guidance for writing, reviewing, and monitoring operations procedures to ensure the content is technically correct and the wording and format are clear and concise.

16.0 DOE Order 433.1, Maintenance Management Program for DOE Nuclear Facilities; and DOE Guide 433.1-1 Nuclear Facility Maintenance Management Program Guide for Use with DOE O 433.1

16.1 DOE Order 433.1, Maintenance Management Program for DOE Nuclear Facilities.

Revision: 1 June 2001
Sponsoring Organization: US Department of Energy, Office of Nuclear Safety

WTP Specific Tailoring

The following tailoring of DOE Order 433.1, *Maintenance Management Program for DOE Nuclear Facilities* is required for use by the WTP contractor as an implementing standard for the preparation of the WTP Maintenance Program.

Section 3, Page 1 APPLICABILITY

Delete section in its entirety.

Justification: The WTP Maintenance Program will follow the requirement section of DOE O 433.1 as tailored below.

Section 4.a, Page 2 REQUIREMENTS

Change 4.a to "A nuclear facility maintenance management program must contain a DOE-approved Maintenance Implementation Plan (MIP) that addresses the following elements using the graded approach."

Justification: DOE O 430.1A is not a standard imposed on the WTP or committed to in an authorization basis document.

Section 4.a(2), Page 2 REQUIREMENTS

Delete section 4.a(2).

Justification: The preventive maintenance program will handle any inspections that are required for the term of this project. Operations will also be providing a surveillance program that will be inspecting equipment and systems. Problems identified with the equipment or systems will then be handled through the work control process.

Section 4.a(9), Page 3 REQUIREMENTS

Delete section 4.a(9).

Justification: DOE M 420.1A is not a standard imposed on the WTP or committed to in an authorization basis document.

Section 4.a(10), Page 3 REQUIREMENTS

Change to "An accurate maintenance history that compiles maintenance, resource, and cost data in a system which is retrievable and capable of entering required-maintenance costs, actuarial maintenance costs, and availability data and failure rates for mission-critical and safety SSCs."

Justification: DOE O 430.1A is not a standard imposed on the WTP or committed to in an authorization basis document.

Section 4.c(1), Page 3 REQUIREMENTS

Change to "the Integrated Safety Management System (ISMS)".

Justification: DOE P 450.4 is not a standard imposed on the WTP or committed to in an authorization basis document.

Section 4.c(2), Page 3 REQUIREMENTS

Delete section 4.c(2).

Justification: DOE O 430.1A is not a standard imposed on the WTP or committed to in an authorization basis document.

Section 4.c(4), Page 3 REQUIREMENTS

Change to "the *Quality Assurance Manual*".

Justification: Clarifies the WTP use and DOE acceptance of the *Quality Assurance Manual*.

Section 4.e, Page 3 REQUIREMENTS

Delete section 4.e.

Justification: The WTP does not have an established maintenance management program under DOE O 4330.4B.

Section 5, Page 4 RESPONSIBILITY

Delete section 5.

Justification: DOE M 411.1-1B is not a standard imposed on the WTP or committed to in an authorization basis document.

Section 6, Page 4 REFERENCES

Delete section 6.

Justification: Not necessary - avoids confusion.

Section 7, Page 6 REFERENCES

Delete section 7.

Justification: Does not apply for WTP use.

Attachment 1 CONTRACTOR REQUIREMENTS DOCUMENT

Delete Attachment 1.

Justification: Avoids confusion with duplication of requirements. The WTP intends to follow the requirement section of DOE O 433.1.

16.2 DOE Guide 433.1-1, Nuclear Facility Maintenance Management Program Guide for Use with DOE O 433.1

Revision: 5 September 2001
Sponsoring Organization: Department of Energy, Office of Nuclear Safety

WTP Specific Tailoring

The following tailoring of DOE Guide 433.1-1, *Nuclear Facility Maintenance Management Program Guide for Use with DOE O 433.1*, is required for use by the RPP-WTP contractor as an Implementing Standard for the preparation of the RPP-WTP Maintenance Program.

Section 4.4.3.2, Page 65 Preventive Maintenance

Replace the text with:

"Predictive maintenance will be integrated into the overall preventive maintenance program so that planned maintenance can be performed prior to equipment failure. Not all equipment conditions and failure modes can be applied. Reliable predictive maintenance will be selectively applied. Reliable predictive maintenance activities involves periodic monitoring in order to forecast component degradation so that (as needed) planned maintenance may be performed prior to equipment failure. Not all equipment conditions and failure modes can be monitored, therefore, predictive maintenance should be selectively applied. In addition, corrective maintenance efficiency may be improved by directing repair efforts (manpower, tooling, and parts) at problems detected using predictive maintenance techniques.

Predictive maintenance will be limited to components and systems that are significantly important to the safe and reliable operation of the plant. The program will collect, trend, and analyze data and initiate planned actions for degrading equipment. The effectiveness of the program is dependent on the accuracy of equipment degradation rate and time to failure assessment."

Justification: Clarification is needed to ensure that the RPP-WTP preventive maintenance program contains all the aspects of preventive maintenance.

17.0 Implementation of Class 1E, IEEE Standards

Introduction:

The following IEEE standards are called out as implementing standards within the SRD, and provide criteria for "Class 1E" equipment and systems for nuclear power generating stations. Since the RPP-WTP project is not a nuclear power generating station, and does not use the term "Class 1E" in the project design documents, the question arises on how these standards will be applied to the RPP-WTP systems and equipment.

Implementing standards for Class 1E systems and equipment:

- IEEE 308-1991 — Criteria For Class 1E Power Systems for Nuclear Power Generating Stations
- IEEE-323-1983 — IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
- IEEE-344-1987(R1993) — Recommended Practice for Seismic Qualification of Class 1E equipment for Nuclear Power generating Stations
- IEEE 384-1992 — Standard Criteria for Independence of Class 1E Equipment and Circuits
- IEEE 628-1987 — Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations.
- IEEE 741-1990 — Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations.

Equipment and systems in nuclear power generating stations are classified as either Class-1E or Non-Class 1E and the design criteria has been clearly defined for each classification. The RPP-WTP implements a defense in depth strategy, with a graded approach to equipment and system safety classification. Therefore there is no clear correlation between the term "Class 1E" and a single safety classification within the RPP-WTP.

The ISM process will also determine the active SDC, SDS or SS equipment and systems that shall be subject to selected design criteria, of the above listed IEEE Class 1E standards. The ISM process will then provide reliability requirements for each control strategy. These reliability requirements determine when control strategies require independence, redundancy, and seismic qualifications.

18.0 IEEE-308, Criteria for Class 1E Power Systems for Nuclear Power Generating Stations

Revision: 1991

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-308 is required for use by the RPP-WTP project as an implementing standard for the SDC/SDS electrical power system design.

Pages 1-21, All Sections Clarification of Nuclear Power Generating Station Terminology

The term "Standby Generator" in the Standard is synonymous with "Emergency Generator" in the RPP-WTP.

Justification: As determined by the ISM review process, the Standby Generators on the RPP-WTP are not classified as SDC while the Emergency Generators are classified as SDC.

The term "Main Control Room" in the Standard is synonymous with the "Respective facility control room" in the RPP-WTP.

Justification: The RPP-WTP does not have a single control room for the entire plant. Each facility has its own control room.

Pages 1-21, All Sections Nuclear Power Generating Station Terminology Not Applicable to the RPP-WTP

The following terminology is not applicable to the RPP-WTP and can be disregarded when encountered in IEEE-308.

- Multi-unit, multi-unit stations or multi-unit nuclear power generating stations
- Reactor, reactor coolant pressure boundary, reactor trip system, or reactor protection system
- fuel cladding

Justification: These terms are specific to nuclear power generating stations and have no equivalent function or term in the RPP-WTP.

Pages 4-5, Section 3.0 References

The following reference standards (and respective footnotes) do not apply for the RPP-WTP.

- [1] C.F.R. (Code of Federal Regulations), Title 10: Energy, Part 100, published by Office of the Federal Register, 1992. (Reactor Site Criteria)

Justification: This document contains criteria for licensing of nuclear power generating stations and doesn't apply for the RPP-WTP. RPP-WTP site criteria are included as part of the SRD.

Pages 4-5, Section 3.0 References, (continued)

The following reference standards (and respective footnotes) do not apply for the RPP-WTP.

- [2] IEEE std-317-1983 (reaff 1988), IEEE Standard for Electric Penetration Assemblies in Containment structures for Nuclear Power Generating Stations, (ANSI).

Justification: Containment electrical penetration assemblies are unique to the containment structure of Nuclear Power Generating Stations and have no equivalent in the RPP-WTP project.

- [9] IEEE Std 415-1986, IEEE Guide for Planning of Pre-Operational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Stations.

Justification: This standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for pre-operational testing programs will be developed internally for the RPP-WTP project.

- [13] IEEE Std 494-1974 (reaff 1990), IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations.

Justification: This standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for identification of documents related to SDC/SDS equipment of will be developed internally for the RPP-WTP project.

- [15] IEEE Std 603-1991, IEEE Criteria for Safety Systems for Nuclear Power Generating Stations.

Justification: This standard will be replaced by ANSI/ISA-S84.01-1996, Per 24590-WTP-ABCN-ESH-01-027.

The following reference Standards shall be included:

- [19] DOE/RL-96-0006, Revision 1, Top-level radiological, nuclear, and process safety standards and principles for TWRS privatization contractors.

Justification: This is a regulatory basis document for the RPP-WTP per the SRD.

- [20] ANSI/ISA-S84.01-1996, Application of Safety Instrument Systems for the Process Industries.

Justification: ANSI/ISA-S84.01-1996, replaces IEEE-603 on the RPP-WTP, Per 24590-WTP-ABCN-ESH-01-027.

The following standards are listed in the SRD with revision dates that are different from the latest revision. The revision of the standard listed in the SRD shall be used for the RPP-WTP.

- [6] IEEE std-379-1994, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Station Safety Systems.
- [7] IEEE std-384-1992, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits, (ANSI).
- [12] IEEE std-485-1983, IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Station and Substations, (ANSI).
- [16] IEEE std-741-1990, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations.

Pages 5-6, Section 4.0, Definitions

- Replace definition of **administrative controls** with the following:

  Provisions relating to organization and management, procedures, record keeping, assessment, and reporting necessary to ensure safe operation of the facility.

Justification: This definition is from Appendix B of the SRD, Volume II.

- Replace the definition of **design basis events** with the following:

  Postulated events providing bounding conditions for establishing the performance requirements of structures, systems, and components that are necessary to: 1) ensure the integrity of the safety boundaries protecting the worker; 2) place and maintain the facility in a safe state indefinitely; or 3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design-Basis Events also establish the performance requirements of the structures, systems and components whose failure under Design-Basis Event conditions could adversely affect any of the above functions."

Justification: This definition is from DOE/RL-96-0006.

- Delete the clause, "of a unit, other that reactor trip or those used for only normal operation" for the definition of **engineered safety features.**

Justification: This clause applies specifically to nuclear power generating stations and is being deleted in order to clarify the definition of the term as it applies to the RPP-WTP.

- Replace the definition of **Nuclear power generation station** with the following:

  The RPP-WTP.

Justification: This substitution clarifies how the term applies to the RPP-WTP.

- Replace the definition of **safety function** with the following:

  "Any function that is necessary to ensure: 1) the integrity of the boundaries retaining the radioactive materials, 2) the capability to place and maintain the facility in a safe state; or 3) the capability to prevent or mitigate the consequences of facility conditions that could result in radiological exposure to the general public or workers in excess of appropriate limits."

Justification: The definition is from DOE/RL-96-0006.

- Replace the definition of **safety system** with the following:

  A SDC/SDS system, as determined in the ISM review process.

  Justification: The standard defines safety system terms of reactor protection which doesn't apply to the RPP-WTP. This definition clarifies what and how a safety system is determined on the RPP-WTP.

- Replace the word "station" with "RPP-WTP" for the definition of **significant.**

  Justification: The term station refers to nuclear power generating station. This substitution clarifies how the term applies to the RPP-WTP.

- Replace the definition of **Unit** with the following:

  The RPP-WTP.

  Justification: The term unit in the standard applies to a nuclear power generating station. This substitution clarifies how the term applies to the RPP-WTP.

The following definitions are applicable for the RPP-WTP:

Safety Design Class, (SDC): The definition for SDC is provided in Safety Criterion 1.0-6.

Safety Design Significant, (SDS): The definition for SDS is provided in Safety Criterion 1.0-6.

Justification: These terms apply to the classification of structures, systems, and components on the RPP-WTP.

Page 7, Section 5.2, Relationship Between the Safety System and Class 1E Power System

Replace with the following:

The SDC/SDS power distribution system shall, as a minimum, meet the criteria called out in this standard and ISA-S84.01-1996, [21]. The SDC/SDS power distribution system will be designed to ensure that the safety systems supported by the SDC/SDS power distribution system will be able to perform their safety functions during and following design basis events.

Justification: Standard ISA-S84.01-1996 is being used in place of IEEE-603, Per 24590-WTP-ABCN-ESH-01-027. The criteria called out in this tailored standard and in ISA-S84.01 are adequate to ensure a reliable SDC/SDS power distribution system.

Page 11, Section 5.6, Location of Indicators and Control

Replace with the following:

SDC/SDS Power distribution system controls will be automated and indication of the safety functions shall be provided in the respective facility control room. Manual control and indication shall be provided outside the facility control rooms.

Justification: The control philosophy for a Nuclear Power Generating Station is not applicable for the RPP-WTP project. A Nuclear Power Generating Station has one main control room, with hard wired controls for all major equipment. The RPP-WTP project has separate control rooms for each facility and automated controls that minimize human factor errors.

Page 11, Section 5.7, Identification

Delete the second sentence.

Justification: IEEE std 494 has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for identification of documents related to SDC/SDS equipment of will be developed internally for the RPP-WTP project.

Page 12, Section 5.13 Circuits That Penetrate Containment.

Not applicable for the RPP-WTP.

Justification: Containment penetration assemblies are unique to the containment structure of Nuclear Power Generating Stations and have no equivalent in the RPP-WTP project.

Page 16, Section 6.4, Instrumentation and Control Power Systems

Replace sub-section 6.4.1 with the following:

The instrumentation and control power systems (ICPS) include power supplies and distribution systems arranged to provide alternating and direct power to the SDC/SDS instrumentation and control, (I&C) loads.

These systems shall be designed to provide highly reliable sources of power to the Programmable Protection System, (PPS) and to SDC/SDS instrumentation and control power systems not integral to the PPS.

Design requirements shall include the following:

1) The SDC/SDS I&C loads shall be distributed between two or more redundant power supplies.
2) The protective actions of each load group shall be independent of the protective action provided by the redundant load groups.
3) An independent direct current power supply shall be provided for each SDC power distribution system load group.
4) Two or more independent alternating current power supplies shall be provided for instrumentation and control.

To accomplish the above requirements, special power supplies may be required that are isolated from the alternating current and direct current power supplies used for normal instrumentation and control of the RPP-WTP.

Justification: This section was re-written to address the I&C requirements of the RPP-WTP.

Page 17, Section 6.5.1, General, (Execute Features)

Delete the last sentence and add the following.

The execute features will be subject to the functional and design requirements in ISA-S84.01, [21] and the requirements called out during the ISM cycle process.

Justification: Standard IEEE-603 is being replaced by ISA-S84.01, Per 24590-WTP-ABCN-ESH-01-027

Page 17, Section 6.5.1, Manual Control

Delete line 3) and replace with the following.

3) Be shown by analysis not to defeat the requirements in ISA-S84.01, [21] as well as the requirements called out during the ISM cycle process.

Justification: Standard IEEE-603 is being replaced by ISA-S84.01, Per 24590-WTP-ABCN-ESH-01-027

Page 20, Section 7.3, Pre-operational System Test

Delete reference to IEEE Std 415.

Justification: This standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for pre-operational testing programs will be developed internally for the RPP-WTP project.

Pages 20-21, Section 8.0, Multiunit Station Considerations

Not applicable to the RPP-WTP.

Justification: This section is specific to Nuclear Power Generating Stations with more that one reactor and has no equivalent application in the RPP-WTP.

19.0 IEEE-384, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits

Revision: 1992

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-384 is required for use by the RPP-WTP project as an implementing standard for SDC, SDS or SS electrical equipment and circuit design.

Pages 1-21, All Sections Clarification of Nuclear Power Generating Station Terminology

The term "Standby Generator" in the Standard is synonymous with "Emergency Generator" in the RPP-WTP.

Justification: As determined by the ISM review process, the Standby Generators on the RPP-WTP are not classified as SDC while the Emergency Generators are classified as SDC.

Page 1, Section 2.0, Purpose

Delete the reference to IEEE-603.

Justification: Standard ISA-S84.01-1996 is being used in place of IEEE-603 per 24590-WTP-ABCN-ESH-01-027, so the reference to IEEE-603 is not applicable.

Page 1, Section 3.0, References

The following reference standards, do not apply for the RPP-WTP.

- [1] ANSI/ANS-58.2-1988, Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture.

Justification: This document is applicable to the high pressure steam lines found in a nuclear power generating stations and doesn't apply for the RPP-WTP.

- [4] ANSI/NFPA 803-1988, Fire Protection for Light Water Nuclear Power Plants.

Justification: This document specifically addresses nuclear power generating stations. Per section 4.5 of volume II of the SRD, the RPP-WTP will use NFPA 801-2003 as an implementing standard for fire protection.

- [11] IEEE Std 494-1974 (reaff 1990), IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations.

Justification: This standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. Procedures for identification of documents related to SDC, SDS or SS equipment will be developed internally for the RPP-WTP project.

- [12] IEEE Std 603-1991, IEEE Standard Criteria for Safety Criteria for Safety Systems for Nuclear Power Generating Stations.

Justification: This standard being replaced by ANSI/ISA-S84.01-1996, Per 24590-WTP-ABCN-ESH-01-027.

The following reference Standards shall be included:

- [16] DOE/RL-96-0006, Revision 1, Top-level radiological, nuclear, and process safety standards and principles for TWRS privatization contractors.

Justification: Called out as a regulatory basis in the SRD.

- [17] ANSI/ISA-S84.01-1996, Application of Safety Instrument Systems for the Process Industries.

Justification: Replaces IEEE-603 per 24590-WTP-ABCN-ESH-01-027.

- [18] NFPA 801-2003, Standard for Fire Protection for Facilities Handling Radioactive Materials

Justification: Called out as an implementing standard under safety criteria 4.5-1 through 4.5-4.

Pages 2-3, Section 4.0, Definitions

The following Change apply to the definitions:

- Replace the note that follows the definition of **associated circuits** with the following:

  Note - Circuits include the interconnecting cabling and connected loads. This definition will apply to circuits meeting the following criteria

  - The only Non-SDC circuits that would be associated with SDC circuits would be those circuits classified as SDS. Such circuits shall meet the criteria called out in section 5.5 of IEEE 384-1992
  - There will not be any non-SDS circuits associated with SDS circuits other than the ones described in the first bullet.
  - There will not be any non-SS circuits associated with SS circuits.
  - SDS circuits associated with SDC circuits shall be subject to the criteria called out in section 5.5 of IEEE 384-1992.

Justification: When circuits for SDS equipment are routed with SDC circuits, the SDS circuits will be treated as associated circuits.

- The definition of **design basis events** shall be replaced with the Following:

"Postulated events providing bounding conditions for establishing the performance requirements of structures, systems, and components that are necessary to: 1) ensure the integrity of the safety boundaries protecting the worker; 2) place and maintain the facility in a safe state indefinitely; or 3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design-Basis Events also establish the performance requirements of the structures, systems and components whose failure under Design-Basis Event conditions could adversely affect any of the above functions."

Justification: This definition is from DOE/RL-96-0006.

The following definitions shall be added.

Safety Design Class, (SDC): The definition for SDC is provided in Safety Criterion 1.0-6.

Safety Design Significant, (SDS): The definition for SDS is provided in Safety Criterion 1.0-6.

Safety Significant, (SS): The definition for SS is provided in Safety Criterion 1.0-6 and Appendix A, Section 6.

Justification: These terms apply to the classification of structures, systems, and components on the RPP-WTP.

Page 3, Section 5.3, Equipment and Circuits Requiring Independence

Replace with the following sentence:

Equipment and circuits requiring independence shall be determined during the ISM review cycle and shall be identified on documents and drawings in a distinctive manner.

Justification: The reference to IEEE-494 is not applicable since this standard has been withdrawn by the IEEE standards committee and no replacement standard has been recommended. This standard is not called out as an implementing standard in the SRD. The ISM process will provide reliability requirements for each control strategy. These reliability requirements determine when control strategies require independence, redundancy, and seismic qualifications.

Page 9, Section 6.1.3.2, Area Boundaries

Replace the reference to NFPA 803-1988[4] with NFPA 801-2003 [19].

Justification: Standard NFPA 803-1998 is not applicable for the RPP-WTP. Per section 4.5 of the SRD, NFPA 801-2003 shall be used for the RPP-WTP.

Page 15, Section 6.5, Containment Electrical Penetrations

Not applicable for the RPP-WTP.

Justification: Containment electrical penetration assemblies are unique to the containment structure of Nuclear Power Generating Stations and have no equivalent in the RPP-WTP project.

20.0 IEEE-338, Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems

Revision: 1987

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-338 is required for use by the RPP-WTP project as an implementing standard for ITS system design and operation.

Page 1, Section 1.0 Scope

Replace the reference to IEEE Std 603-1980 [4], and replace with ANSI/ISA 84.01-1996 [6].

Justification: ANSI/ISA-S84.01-1996, replaces IEEE-603 on the RPP-WTP, Per 24590-WTP-ABCN-ESH-01-027.

Pages 1-2, Section 2.1, Definitions

Replace the definition for "safety function" with the following:

> **Safety Function.** "Any function that is necessary to ensure: 1) the integrity of the boundaries retaining the radioactive materials, 2) the capability to place and maintain the facility in a safe state; or 3) the capability to prevent or mitigate the consequences of facility conditions that could result in radiological exposure to the general public or workers in excess of appropriate limits."

Justification: The definition is from DOE/RL-96-0006.

Pages 2-3 , Section 3.0 References

The following reference standards (and respective footnotes) do not apply for the RPP-WTP.

- [4] IEEE Std-603-1980, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations

Justification: This standard being replaced by ANSI/ISA-S84.01-1996, Per 24590-WTP-ABCN-ESH-01-027.

- [5] ANSI/IEEE/ANS 7-4.3.2-1982, Application Criteria for Programmable Digital Computer systems in Safety Systems of Nuclear Power Generating Stations

Justification: This standard applies for computer systems used in nuclear power generating stations and, due to the rapid advances in computer designs, is out of date for use on the RPP-WTP. ANSI/ISA 84.01-1996, will be used on the RPP-WTP in place of this standard.

The following reference Standard shall be included:

- [6] ANSI/ISA 84.01-1996, Application of Safety Instrumented Systems for Process Industries

Justification: ANSI/ISA-S84.01-1996, replaces IEEE-603 on the RPP-WTP, Per 24590-WTP-ABCN-ESH-01-027.

- [7] DOE/RL-96-0006, Revision 1, Top-level radiological, nuclear, and process safety standards and principles for TWRS privatization contractors.

Justification: This is a regulatory basis document for the RPP-WTP per the SRD.

The following reference Standard revision shall be used in compliance with the SRD:

- [3] IEEE Std 308-1991, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations

Justification: This revision date referenced in the SRD for this standard shall be used for the RPP-WTP.

Page 4, Section 5.0 Design Requirements

For paragraph number 7: The term "main control room" shall mean the respective facility control room for the RPP-WTP project.

Justification: The RPP-WTP project does not have a single control room like a nuclear power generating station. Each facility has its own control room.

Page 5, Section 6.1 General Consideration

For paragraph number 2: Replace the term "reactor operation" with "system operation".

Justification: The term "reactor operation" is specific to a nuclear power generating station.

21.0 IEEE-628, IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations

Revision: 1987

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-628 is required for use by the RPP-WTP project as an implementing standard for the SDC, SDS or SS raceway design, installation, and qualification.

Pages 1-2, Section 3.0 Definitions

The following definitions shall be included:

Safe Shutdown Earthquake (SSE): A design basis earthquake for RPP-WTP and the applicability to systems, structures and components, (SSCs). Criteria for this event is contained in 24590-WTP-SRD-ESH-01-001-01, Safety Requirements Document, (SRD) Volume II, Safety Criterion 4.1-3.

Justification: The above listed definitions were added to help define the applicability of this standard to the RPP-WTP project.

Safety Design Class, (SDC): The definition for SDC is provided in Safety Criterion 1.0-6.

Safety Design Significant, (SDS): The definition for SDS is provided in Safety Criterion 1.0-6.

Safety Significant, (SS): The definition for SS is provided in Safety Criterion 1.0-6 and Appendix A, Section 6.

Justification: The above listed definitions were added to help define the applicability of this standard to the RPP-WTP project.

Pages 4-5, Section 4.0 References

The first sentence shall read.

This standard shall be used in conjunction with the latest version of the following standards. If the referenced standard is listed in the SRD as an implementing standard, then the version of the standard listed in the SRD shall be used.

The following reference standard does not apply for the RPP-WTP.

- [12] IEEE std-634-1978, IEEE Standard Cable Penetration Fire Stop Qualification Test.

Justification: This standard has been withdrawn. Per section 4.5 of the SRD, the implementing fire protection standard for the RPP-WTP will be NFPA 801-2003. Fire stop qualification tests shall be per the Factory Mutual standards.

The following reference Standard shall be included:

- [34] NFPA 801-2003, Standard for Facilities Handling Radioactive Materials

Justification: Called out as an implementing standard per safety criteria 4.5-1 through 4.5-4.

The following standards are listed in the SRD with revision dates that are different from the revisions dates listed in the standard. The following revisions of the below standards shall be used in place of the revisions referenced in the body IEEE-628.

- [4] ANSI/ACI 349-97, Code Requirements for Nuclear Safety-Related Concrete Structures
- [6] ANSI/ASME NQA-1-1989 Quality Assurance Program Requirements for Nuclear Facilities.
- [10] IEEE std-344-1987, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.
- [11] IEEE std-384-1992, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits, (ANSI).
- [19] NFPA70-1999, National Electric code, (Note: per 24590-WTP-ABCN-ESH-01-025)

Page 4, Section 5.1, General, (Design)

Delete the seventh paragraph and replace with the following:

Raceways that penetrate a fire barrier shall have fire stops installed in accordance with NFPA 801-2003, [34]. Fire stops will utilize UL-listed and/or Factory Mutual-approved assemblies with a fire rating equal to or greater than the rating of the fire barrier.

Justification: IEEE 634-1978 has been withdrawn. Since IEEE-690 references IEEE-634, it was deleted from the paragraph as well. NFPA 801 is an implementing standard for fire protection in the RPP-WTP, per the SRD. The qualification of fire stops for the RPP-WTP will be addressed internally by the fire protection group.

Page 5, Section 5.6, Environmental Consideration

Delete second paragraph.

Justification: The requirement for raceway systems installed in the containment is specific to nuclear power generating stations and does not have an equivalent application to the RPP-WTP.

Page 11, Section 5.10.1.1.5, Operating Basis Earthquake, (OBE) Loads

This section is not applicable to the RPP-WTP

Justification: OBE loads have been determined to be not applicable to the RPP-WTP, refer to 24590-WTP-ABCN-ESH-01-013.

Page 11, Section 5.10.1.2, Load Combinations

This OBE and SRV are not applicable to the RPP-WTP

Justification: OBE loads have been determined to be not applicable to the RPP-WTP plant, refer to 24590-WTP-ABCN-ESH-01-013. As stated in section 5.10.1.1.4, SRV loads only apply to BWR nuclear power generating stations and therefore do not apply to the RPP-WTP

22.0 IEEE-344, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

Revision: 1987(R1993)

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-344 is required for use by the RPP-WTP project as an implementing standard for SDC/SDS electrical and instrument system design.

Pages 1-43, All Sections Clarification of Nuclear Power Generating Station Terminology

The term "Class 1E" in the Standard applies to "SC-I SDC" in the RPP-WTP.

Justification: The Scope, section 1.0, of IEEE-344 applies to equipment that needs to function during and after an SSE for a Nuclear Power Generating Station. For RPP-WTP the equipment that needs to function during and after a design basis earthquake is SDC equipment which must be qualified to SC1.

Page 1, Section 1.2 References

Delete reference [5] CFR (Code of Federal Regulations), Title 10: Energy, Part 100, Reactor Site Criteria, published by office of the Federal Register, 1992.

Justification: Reference [5] contains radiation dose criteria and seismic criteria for Nuclear Power Generating Stations and is not applicable to the RPP-WTP project. The applicable criteria for RPP-WTP is found in 24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document (SRD) Volume II, Safety Criteria 2.0-1 for radiological dose and 2.0-2 for chemical hazards. The applicable seismic criteria is contained in 24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document (SRD) Volume II, in section 4.1 General Design, Safety Criterion 4.1-3. This Safety Criterion defines Seismic Classes (SC) I, II and III and provides seismic loads and source documents.

Delete reference [3] ANSI/IEEE Std 382-1985, *IEEE Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants.*

Justification: This standard will be replaced with IEEE Std 382-1996. The IEEE Std 382-1996 includes a Required Input Motion (RIM) curve.

Pages 1-2, Section 2. Definitions

Delete the definitions for **Operating basis earthquake (OBE)** and **safe shutdown earthquake (SSE)**.

Add a definition for **design basis earthquake** as: Earthquakes for RPP-WTP and the applicability to systems, structures and components (SSCs) is contained in 24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document (SRD) Volume II, in section 4.1 General Design, Safety Criterion 4.1-3. This Safety Criterion defines Seismic Classes (SC) I, II and III and provide seismic loads and source documents.

Justification: The definition of OBE and SSE are applicable to Nuclear Power Generating Stations and the new definitions is applicable to the RPP-WTP project as defined in the SRD. This is consistent with the tailoring of AISC N690 as documented in ABCN-013.

Pages 1-43, All Sections Clarification of OBE and SSE

The term SSE in the standard is treated as a design basis earthquake. The requirement to apply and document the loads of a number of OBEs before an SSE is deleted from the standard.

Justification: The earthquake applicable to RPP-WTP is the design basis earthquake. The requirement to subject equipment to several OBEs prior to an SSE is not included in the requirements of the SRD for the RPP-WTP project. This is consistent with the tailoring of AISC N690 as documented in ABCN-013.

Page 13, Section 7.1.3.2, Repairs

In the fifth line delete the words, ", such as LOCA,".

Justification: LOCA is a term specific to Nuclear Power Generating Stations and not to the RPP-WTP project.

Page 15, Section 7.1.5, Vibrational Aging

In the last paragraph change the first sentence to read, "The purpose of the vibrational aging is to show that the lower levels of normal and transient vibration associated with plant operation will not adversely affect an equipment's performance of its safety function nor cause any condition to exist that, if undetected, would cause failure of such performance during a subsequent design basis earthquake.

Justification: This sentence within the standard included additional vibration aging of an OBE, but used the terms "lower intensity earthquake" rather than OBE. The rewording is needed to clarify the meaning of the sentence. The requirement to subject equipment to several OBEs prior to an SSE is not included in the requirements of the SRD for the RPP-WTP project. The earthquake applicable to RPP-WTP is the design basis earthquake. This is consistent with the tailoring of AISC N690 as documented in ABCN-013.

Page 16, Section 7.1.6.1, Hydrodynamic Loads

Delete the words, "and the loss-of-coolant accident (LOCA)

Justification: LOCA is a term specific to Nuclear Power Generating Stations and not to the RPP-WTP project.

23.0 IEEE-323, Qualifying Class 1E Equipment for Nuclear Power Generating Stations

Revision: 1983

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-323 is required for use by the RPP-WTP project as an implementing standard for ITS electrical and instrument system design.

Pages 1-2, Section 2, References

The following reference Standard shall be included:

- [20] DOE/RL-96-0006, Revision 1, Top-level radiological, nuclear, and process safety standards and principles for TWRS privatization contractors.

Justification: The added references are applicable for the RPP-WTP project.

Pages 2-3, Section 3, Definitions

- Modify the definition of **harsh environment** to be: An environment expected as the result of the postulated service condition appropriate for the design basis event of the RPP-WTP. It is an environment that exceeds the conditions of a mild environment. Equipment that do not experience an environment beyond a mild environment during a design basis event can be considered to be in a mild environment.

Justification: A **harsh environment,** as defined by this standard, applies to a Nuclear Power Generating Station and are the result of a loss of cooling accident (LOCA)/high energy line brake (HELB) inside the containment and post-LOCA or HELB outside containment. The modified definition applies to RPP-WTP.

This modified definition is further supported by 10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, which states, in section C: " Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences."

The definition of **mild environment** within the standard states:

"An environment expected as a result of normal service conditions and extremes (abnormal) in service conditions where seismic is the only design basis event (DBE) of consequences."

Therefore the normal operating environment for a SSC is considered a "mild environment" by this definition.

The following definition is applicable for the RPP-WTP:

- The definition of **design basis events** shall be added with the definition from DOE/RL-96-0006, which states:

  "Postulated events providing bounding conditions for establishing the performance requirements of structures, systems, and components that are necessary to: 1) ensure the integrity of the safety boundaries protecting the worker; 2) place and maintain the facility in a safe state indefinitely; or 3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design-Basis Events also establish the performance requirements of the structures, systems and components whose failure under Design-Basis Event conditions could adversely affect any of the above functions."

Justification: The above listed definition was added to be applicable to the RPP-WTP project.

Page 14-15, Section 7, Simulated Test Profiles

Delete this section.

Justification: This section is specific to Nuclear Power Generating Stations and describes profiles and margin for LOCA/HELB harsh environments.

24.0 IEEE-379, Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems

Revision: 1994

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

RPP-WTP Specific Tailoring

The following tailoring of IEEE-379 is required for use by the RPP-WTP project as an implementing standard for SDC/SDS system design and operation.

Page 1 , Section 1.1 Scope

Rewrite the scope to read: This document covers the application of the single-failure criterion to the electrical power instrumentation, and control portions of facility safety systems as determined by the ISM Process.

Justification: Application of IEEE-379 to the RPP-WTP project is determined by the ISM Process.

Page 1 , Section 1.2 Purpose

Remove the second paragraph.

Justification: IEEE 603 is not used for WTP project. See ABCN 24590-WTP-ABCN-ESH-01-027.

Pages 1-2 , Section 2.0 References

The following reference standards (and respective footnotes) do not apply for the RPP-WTP.

- IEEE Std-603-1980, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations

The following reference Standard shall be included:

- ANSI/ISA 84.01-1996, Application of Safety Instrumented Systems for Process Industries
- DOE/RL-96-0006, Revision 1, Top-level radiological, nuclear, and process safety standards and principles for TWRS privatization contractors.

Justification: Reference IEEE 603 is not used for the WTP project. See ABCN 24590-WTP-ABCN-ESH-01-027. ANSI/ISA 84.01 and DOE/RL-96-0006 are used for the design and implementation of safety systems for the WTP project.

Pages 1-2, Section 2.1, Definitions

For WTP, the definitions for the following is contained in DOE/RL-96-0006.

<u>Common-Cause Failure.</u> Dependent failures that are caused by a condition external to a system or set of components that make system or multiple component failures more probable than multiple independent failures.

Design Basis Events. Postulated events providing bounding conditions for establishing the performance requirements of structures, systems, and components that are necessary to: 1) ensure the integrity of the safety boundaries protecting the worker; 2) place and maintain the facility in a safe state indefinitely; or 3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design Basis Events also establish the performance requirements of the structures, systems, and components whose failure under Design Basis Event conditions could adversely affect any of the above functions.

Safety Function. "Any function that is necessary to ensure: 1) the integrity of the boundaries retaining the radioactive materials, 2) the capability to place and maintain the facility in a safe state; or 3) the capability to prevent or mitigate the consequences of facility conditions that could result in radiological exposure to the general public or workers in excess of appropriate limits."

Justification: The definition is from DOE/RL-96-0006.

Page 5, Section 5.6 **Shared Systems**

Remove section 5.6

Justification: The WTP project does not have shared systems. This applies to Nuclear Power Generating Stations with multiple units.

Page 5, Section 6.1 **Procedure**

For items 1-3, remove examples from the text.

Justification: These examples are unique to Nuclear reactors and do not contribute to the understanding of the standard for use in the WTP project.

25.0 NUREG-0800, Standard Review Plan, Section 6.4, "Control Room Habitability System", Section II

Revision: Draft Revision 3, April 1996
Sponsoring Organization: US Nuclear Regulatory Commission

WTP Specific Tailoring

The following tailoring of NUREG-0800 is required for use by the WTP project as an implementing standard for control room habitability.

Pages 6.4-4 through 6.4-6, All Items

Remove all redline, strikeout, and change annotations from the original Draft NUREG text.

Justification: Removal of the redline, strikeout, and annotations from the NRC draft is necessary to avoid confusion between text changed as WTP tailoring versus text altered as part of the NRC draft.

Pages 6.4-4 through 6.4-6, All Items

Replace all instances of the word "should" with the word "shall".

Justification: The NUREG was a guidance document for NRC licensees, and as guidance the word "should" is appropriate; however, since it is being adopted as a standard the word "shall" is more appropriate.

Page 6.4-4, Item 1 Control Room Emergency Zone

In the title and first sentence change "emergency" to "ventilation".

Justification: For project purposes "control room emergency zone" equates to ventilation zone. Since the word "ventilation" conveys the clearer meaning, the word was changed to avoid confusion.

In Item 1.a, replace the words "... the plant, i.e., the control room, including the critical document reference file." with "...including those vital records necessary to establish and maintain a safe state of the facility;"

Justification: The term "critical reference file" refers to an NRC requirement which does not have an exact equivalent within DOE. The requirement to establish and maintain a vital records program is contained in DOE/RL-94-02, Hanford Emerency Plan, Section 14.3.5.

Page 6.4-4, Item 2 Ventilation System Criteria

Item 2.a, in the third sentence add the words "be determined by safety analysis and" following "shall".

Justification: This wording is more consistent with the use of the SAR results for the WTP.

Item 2.b, delete the second sentence.

Justification: This sentence is deleted because the term "active components" as they refer to concept of "single failure" have been defined elsewhere in project documentation.

Item 2.b, delete the third sentence.

Justification: This sentence referred the reader to an appendix containing an alternative for meeting the criteria cited. The alternative is intended for cases where complex valve or damper configurations have been required to meet the single failure criterion. Credit is allowed for an alternative system that allows a failed valve to be manually repositioned so that it will not interfere with the operation of the system. However, the standby emergency ventilation system planned for the MCR is not a complex system and this alternative is unnecessary. Therefore, this sentence has been deleted.

Pages 6.4-4 through 6.4-5, Item 3 Pressurized Systems

In the first sentence add the words "one of" between "meet" and "the following requirements".

Justification: This change was made to make it clear that based on the type of pressurization system chosen <u>one</u> of the criteria below applied.

In Items 3a and 3c reword the parenthetical phrase "(every 18 months)" to (not to exceed 18 months).

Justification: The frequency of the periodic verification will be determined as part of the SAR process, but will not exceed the 18 month period specified by the NUREG.

In Items 3b and 3c, at the end of the first sentence of 3b in the parentheses "(1)" change to "(a)", and in the first sentence of 3c in the parentheses "(2)" change to "(b)".

Justification: These are typographical errors that existed in the original Draft Revision 3.

In the second sentence of Item 3b, replace the words "at the CP, combined license (COL), or standard design certification stage" with "during system design".

Justification: The deleted words were references to stages in the NRC licensing process which do not apply to the WTP. They were replaced with a term which does apply to the WTP.

Page 6.4-5, Item 4 Emergency Standby Atmospheric Filtration System

Delete the first two sentences.

Justification: These sentences have been deleted since the quantity of radioactive iodine in the waste to be processed is very small and under accident conditions does not pose a significant airborne hazard as it does for commercial nuclear power facilities (24590-PTF-M4C-V11T-00003, Rev 1).

At the end of the third sentence change "(Reference 14)" to "(Reference 1)".

Justification: In the revised reference list the reference for the ASME Code is number 1.

In the fourth sentence add the words "The evaluation of" at the beginning of the sentence, replace "chlorine or other toxic gases" with "hazardous chemicals shall be consistent with the methodologies presented", delete "is addressed", add "(Reference 2)" following 1.78, and replace "1.95" with "Draft Regulatory Guide DG-1111".

Justification: The fourth sentence has been edited to indicate that the process used by the project to evaluate the control room habitability will be consistent with that contained in the NRC guidance documents cited in the sentence. The reference to Regulatory Guide 1.95 was deleted because the latest revision of Regulatory Guide 1.78 (Rev 1) now incorporates this guide and RG 1.95 has been withdrawn by the NRC. Draft Regulatory Guide DG-1111 was added because it contains the latest guidance on modeling atmospheric dispersion for evaluating control room habitability and should be considered. The words "chlorine or other toxic gases" were deleted and the words "hazardous chemicals" were added to be consistent with the new title of Regulatory Guide 1.78, *Evaluating Habitability for a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.*

Add the following sentences at the end: "Exposure thresholds for protection of control room personnel from radiological and chemical hazards are provided in the WTP *Safety Requirements Document Volume II*, Safety Criterion 4.3-7. Evaluation results will be compared to these exposure thresholds to ensure that the control room emergency standby atmospheric filtration system is capable of maintaining personnel protection during off-normal and emergency events."

Justification: These sentences have been added to specifically call out the exposure thresholds for control room personnel specified in the SRD. This was done because the toxic limits used in the regulatory guides cited do not match those called for in *Safety Requirements Document Volume II*, Safety Criterion 4.3-7. And to clarify that the exposure thresholds cited in the SRD are to be compaerd with the evaluation results to ensure that adequate protection is provided for control room personnel.

Page 6.4-5, Item 5 Relative Location of Source and Control Room

In Item 5.a, second sentence, replace the word "dose" with "safety" and delete "(Ref. 9)".

Justification: The word "dose" in the second sentence was changed to "safety" to reflect the safety analysis process which will provide the analysis on which to base the location of the control room intakes. "(Ref. 9)" was deleted from the end of the sentence to eliminate a reference to a 1974 document. The guidance provided by this document has been superseded by recent revisions to the NRC Regulatory Guides and newly issued Draft Regulatory Guides.

In Item 5.b, second sentence, replace the words "The acceptance criteria for the" with "The evaluation of", replace "system are provided in the regulatory positions of Regulatory Guide 1.78 with respect to postulated hazardous chemical releases in general and in Regulatory Guide 1.95 with respect to accidental chlorine releases in particular" with "during the postulated release of hazardous chemicals shall be consistent with the methodologies presented in Regulatory Guide 1.78 (Reference 2) and Draft Regulatory Guide DG-1111 (Reference 3)".

Justification: The second sentence has been edited to indicate that the process used by the project to evaluate control room habitability during a postulated release of hazardous chemicals will be consistent with that contained in the NRC guidance documents cited in the replacement words.

Add the following final sentence: "Exposure thresholds for the evaluation of control room habitability are provided in the WTP *Safety Requirements Document Volume II*, Safety Criterion 4.3-7."

Justification: The last sentence was added to specify that the exposure thresholds called out in the *Safety Requirements Document Volume II*, Safety Criterion 4.3-7 are to be used in this evaluation.

Pages 6.4-20 through 6.4-21 References

Delete reference numbers 1, 2, 3, 4, 5, 10, 11, 12, and 13.

Justification: Remove those references which are not used in the portion of NUREG-0800 cited in the *Safety Requirements Document Volume II.*

Delete reference number 6.

Justification: See justification for deleting reference to Regulatory Guide 1.52 under Page 6.4-5, Item 4 above.

Change reference number 7 to reflect the update of Regulatory Guide 1.78 to Revision 1 and renumber to be reference 2.

Justification: See justification presented under Page 6.4-5, Item 4 above.

Delete reference number 8.

Justification: See justification presented under Page 6.4-5, Item 4 above.

Delete reference number 9.

Justification: See justification presented under Page 6.4-5, Item 5.a above.

Renumber reference 14 to be reference 1.

Justification: The revised reference list has been reordered based on the order in which the references appear in the tailored implementing standard.

Add reference 3: "USNRC, *Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants*, Draft Regulatory Guide DG-1111, December 2001."

Justification: This reference was added as a result of the tailoring process; see justification under Page 6.4-5, Item 4 and Page 6.4-6, Item 5 above.

26.0 ASME B31.3-1996, Process Piping

Revision: 1996
Sponsoring Organization: ASME

WTP Specific Tailoring

The following tailoring of ASME B31.3, *Process Piping*, is required for use by the WTP contractor as an Implementing Standard for the fabrication and installation of those portions of the C5V ductwork that are being embedded in concrete and for the use of ASME B16.9 welding tees in accordance with ASME B31.3-2002. The tailored sections of ASME B31.3 applicable to embedded ductwork will only be utilized to the extent that it will cover the fabrication, installation, and inspection (and associated testing) of Category D fluid service piping being used as C5 ductwork. Air Testing requirements for this ductwork will be compliant with ASME AG-1. Below is a description of those portions of ASME B31.3 that apply to fabrication, installation, and inspection of Category D fluid service piping and the sections of the SRD that they will apply to. The tailored sections of ASME B31.3 applicable to welding tees will only be used for ASME B16.9 welding tees. As long as the stress intensification factors from ASME B31.3-2002 are used in the stress analysis for the welding tees, welding tees fabricated to either the 1996 or the 2002 edition of ASME B31.3 can be used. Below is a description of those portions of ASME B31.3, Appendix D, Table D300, that apply to welding tees and the section of the SRD to which they will apply.

SRD 4.4-3 will comply with the following sections of ASME B31.3-1996, *Process Piping*. These sections of ASME B31.3 are applicable for embedded ductwork.

Chapter 3, Materials

Chapter 5, Fabrication

~~Table 341.3.2, Visual acceptance criteria for Category D fluid service piping~~

Justification: Due to wall thickness requirements of duct embedded in concrete, piping materials are required. ASME B31.3 will apply to materials, fabrication, and inspection standards as appropriate. Testing requirements for nuclear air treatment systems will be consistent with ASME AG-1.

SRD 5.1-2 will comply with the following sections of ASME B31.3-1996, *Process Piping*. These sections of ASME B31.3 are applicable for embedded ductwork.

Chapter 3, Materials

Chapter 5, Fabrication

Table 341.3.2, Visual acceptance criteria for Category D fluid service piping

Justification: Due to wall thickness requirements of duct embedded in concrete, piping materials are required. ASME B31.3 will apply to materials, fabrication, and inspection standards as appropriate. Testing requirements for nuclear air treatment systems will be consistent with ASME AG-1.

Piping providing a confinement function in accordance with SRD 4.2-2 will comply with ASME B31.3-1996, Process Piping, with the following modification:

In Table D300, the description of welding tee per ASME B16.9 shall be revised so it is consistent with that shown in Table D300 of ASME B31.3-2002:

Description	Flexibility Factor k	Stress Intensification Factor [Notes (2), (3)] Out-of-Plane, i_0	In-Plane i_i	Flexibility Characteristic, h	Sketch
Welded tee per ASME B16.9 [Notes (2), (4), (6), (11), (13)]	1	$\frac{0.9}{h^{2/3}}$	$3/4\, i_0 + 1/4$	$3.1\frac{\overline{T}}{r_2}$	Same as ASME B31.3-1996

This means that for welding tees per ASME B16.9, note 11 in Table D300 is also changed to:

(11) If $r_x \geq 1/8 D_b$ and $T_c \geq 1.5\overline{T}$, a flexibility characteristic of $4.4\overline{T}/r_2$ may be used.

Justification: The use of a lower flexibility characteristic for welding tees per ASME B.16.9 in accordance with ASME B31.3-2002 will increase both the out-of-plane and in-plane stress intensification factors. The increased stress intensification factors will reduce the allowable out-of-plane and in-plane moments that can be applied to the welding tee and keep the calculated stress below the stressess allowabled by ASME B31.3-1996.

27.0 DOE Guide 421.1-2, Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830

Revision: 24 October 2001
Sponsoring Organization: Department of Energy, Office of Nuclear and Facility Safety Policy

WTP Specific Tailoring

The following tailoring of DOE Guide 421.1-2, *Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*, is required for use by the WTP contractor as an implementing standard for the preparation of the WTP Safety Analysis Reports.

Throughout

Use of the terms "Documented Safety Analysis" or "DSA" is understood to mean "Final Safety Analysis Report" or "FSAR" for the WTP project.

Justification: The general DSA term used in section 4.1.3 of the guide is interpreted to apply to the Final Safety Analysis Report (FSAR) documentation that will be used to describe the WTP safety analysis.

Section 4.1.3, Page 15 Annual DSA Updates (830.202)

In the 5th paragraph change the last sentence to "However, at least those implemented six months or more before the submittal of the annual update shall be included."

Justification: Changed for consistency with Safety Criterion 9.1-4.

28.0 DOE Order 5480.20A, Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities - Attachment 2, References and Definitions

Revision: 15 November 1994 (Chg1: 12 July 2001)
Sponsoring Organization: US Department of Energy; Office of Environmental, Safety, and Health

WTP Specific Tailoring

The following tailoring of DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements*, Attachment 2, "References and Definitions", is required for use by the WTP contractor as an implementing standard for the preparation of the WTP Project Training Program.

Attachment 2, Chapter I, Page I-6, Section 7.b(1), 2nd paragraph, 2nd sentence

Delete "or in the DOE Guidelines for Job and Task Analysis for Department of Energy Nuclear Facilities, DOE/EP-0095".

Justification: DOE/EP-0095, *DOE Guidelines for Job and Task Analysis for Department of Energy Nuclear Facilities* is not invoked by the WTP contract (DE-AC27-01RV14136).

Attachment 2, Chapter I, Page I-9, Section 7.d, 1st paragraph, 2nd sentence

Delete the 2nd sentence.

Justification: DOE-STD-1060-93 is not invoked by the WTP contract.

Attachment 2, Chapter I, Page I-11, Section 7.e(1)(c)

Delete (c)1: "Training program content shall be in accordance with DOE/EH-0256T, *Radiological Control Manual*, Chapter 6, Training and Qualification."

Justification: DOE/EH-0256T is not invoked by the WTP contract.

Attachment 2, Chapter I, Page I-11, Section 7.e(1)(i)

Delete (i)1: "Training program content shall be in accordance with ANSI/ANS 8.20-1991, *Criticality Safety Training*."

Justification: ANSI/ANS 8.20-1991is not invoked by the WTP contract.

Attachment 2, Chapter I, Page I-13, Section 7.g(3)

Delete (c)1: "Training program content shall be in accordance with DOE/EH-0256T, *Radiological Control Manual*, Chapter 6, Training and Qualification."

Justification: DOE/EH-0256T is not invoked by the WTP contract.

Attachment 2, Chapter I, Page I-19, Section 15, 1st paragraph, 2nd sentence

Delete 2nd sentence: "The guidance in the *Nuclear Information and Records Management Association Guidelines for Management of Nuclear Related Training Records*, TG-17 should be used to help standardize identification, handling, and storage of training records."

Justification: TG-17 is not invoked by the WTP contract. ASME NQA-1-1989 has been identified as the implementing standard for WTP documents and records in accordance with *Safety Requirements Document Volume II*, Safety Criterion 7.3-1.

Attachment 2, Chapter I, page I-20, Section 15.b, 2nd sentence

Replace "DOE 1324.2A, RECORDS DISPOSITION" with "ASME NQA-1-1989, Section 3S-1, 7, "Documentation and Records (including associated supplements)" and ASME NQA-1-1989, Section 3S-1, 17, "Quality Assurance Records (including associated supplements)".

Justification: DOE 1324.2A is not invoked by the WTP contract. ASME NQA-1-1989 has been identified as the implementing standard for WTP documents and records in accordance with *Safety Requirements Document Volume II*, Safety Criterion 7.3-1.

Attachment 2, Chapter II, pages II-1 through II-18

This chapter, in its entirety, is not used.

Justification: Use of this chapter is not applicable to the WTP, as there are no Category A Reactors associated with the project.

Attachment 2, Chapter III, pages III-1 through III-8

This chapter, in its entirety, is not used.

Justification: Use of this chapter is not applicable to the WTP, as there are no Category B Reactors associated with the project.

29.0 DOE Order 420.1A, Facility Safety

Revision: 20 May 2002
Sponsoring Organization: US Department of Energy, Office of Environment, Safety, and Health

WTP Specific Tailoring

The following tailoring of DOE Order 420.1A, *Facility Safety*, is required for use by the WTP contractor as an implementing standard for fire safety.

Page 3, Section 4, Requirements

Change the last sentence of the 1st paragraph to:

> "All new construction shall, as a minimum, conform to the Model Building Codes (i.e., 2000 International Building Code (IBC)) applicable for the state or region, supplemented in a graded manner with additional safety requirements associated with the hazards in the facility."

Justification: This tailoring is necessary to make the use of DOE O 420.1A for the WTP Project consistent with the approved use of the non-structural portions of the 2000 edition International Building Code (IBC) in lieu of the similar portions of the 1997 edition of the Uniform Building Code (UBC). See ABAR 24590-WTP-ABAR-ESH-02-033, Rev 0 approved by OSR Letter 03-OSR-0145 (CCN 054986).

Page 8, Section 4.2.2 Fire Protection Design Requirements

In Item 3 add the following words at the end of the paragraph:

> "In meeting the requirements for fully sprinklered facilities, automatic fire extinguishing systems are not required in the High Level Waste building's high radiation areas containing low combustible loading as identified in Appendix K."

Justification: Any fire in the areas would be small and contained close to the point of origin with minimal radiological consequences. Installation of automatic fire suppression systems in high radiation areas with low combustible loading is not required to reach or maintain safe state. The benefits of installing this system are outweighed by safety concerns associated with having automatic fire suppression systems in these areas. These concerns include the potential of inadvertent actuation resulting in the spread of contamination and impacts to the facility structure from flooding. Actuation of the system would require an operator to authorize the system to be turned off, but since these areas are inaccessible, there would be no practical means to verify the reason for the actuation, and to allow restoration to an operable status.

Page 8, Section 4.2.2 Fire Protection Design Requirements

In Item 4 add the following words to the end of the second sentence:

> "...except this separation shall not be required in rooms where redundant Safety Design Class (for the protection of the public only) systems converge at a common component provided that, to the extent practical, the routing of redundant safety class circuits complies with the physial separation requirements of IEEE Standard 384-1992. If the redundant Safety Design Class system is subject to loss due to a fire event then additional fire protection measures shall be taken to ensure that the redundant Safety Design Class system or component perform its intended safety function."

Justification: The means to separate certain systems or portions thereof into separate fire areas is not possible in some instances due to the nature of the system design. For example, a single tank, which may require constant redundant level indication, is effectively impossible to separate into two fire areas. Areas where this type of situation occurs are exclusively found in C5/R5 areas. Fire hazards analysis will confirm that said systems or components are not subject to fire loss. If fire hazards analysis determines that a common mode failure is possible then additional fire protection measures will be taken to ensure that each SDC systems or component affected will perform its intended safety function.

30.0 IEEE-382, IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies With Safety-Related Functions for Nuclear Power Plants

Revision: 1996

Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

WTP Specific Tailoring

The following tailoring of IEEE-382 is required for use by the RPP-WTP project as an implementing standard for SDC electrical and instrument system design.

Pages 1-32; All Sections **Clarification of Nuclear Power Generating Station Terminology**

The term "nuclear plant", "nuclear power generating stations", and "conventional plant" will be taken to mean the WTP.

Justification: Clarifies how the standard will apply to the WTP project.

Pages 1-32; All Sections **Clarification of OBE and SSE**

The term SSE in the standard is treated as a design basis earthquake. The requirement to apply and document the loads of a number of OBEs before an SSE is deleted from the standard.

Justification: The earthquake applicable to RPP-WTP is the design basis earthquake. The requirement to subject equipment to several OBEs prior to an SSE is not included in the requirements of the SRD for the RPP-WTP project. This is consistent with the tailoring of AISC N690 as documented in 24590-WTP-ABCN-ESH-01-013.

Page 1; Section 1.1 **Scope**

Revise section 1.1 as follows:

This standard describes the qualification of valve actuators and in-line mounted instruments for safety-related functions in nuclear power generating stations.

Justification: IEEE standard 382-1996 provides testing guidance and performance requirements for actuators for power-operated valve assemblies. Current industry practice applies these testing requirements to both actuators and in-line mounted instruments.

Page 1; Section 2 References

Delete reference Code of Federal Regulations, Title 10, Energy - Nuclear Regulatory Commission, Part 100, Jan. 1996.

Justification: Reference contains radiation dose criteria and seismic criteria for Nuclear Power Generating Stations and is not applicable to the RPP-WTP project. The applicable criteria for RPP-WTP is found in 24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document (SRD) Volume II, Safety Criteria 2.0-1 for radiological dose and 2.0-2 for chemical hazards. The applicable seismic criteria is contained in 24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document (SRD) Volume II, in section 4.1 General Design, Safety Criterion 4.1-3. This Safety Criterion defines Seismic Classes (SC) I, II, and III and provides seismic loads and source documents.

Pages 1-3; Section 3 *Definitions*

Delete all definitions, including operating basis earthquake (OBE) and safe shutdown earthquake (SSE), except 3.16 required input motion (RIM).

Add a definition for a design basis earthquake as:

Earthquakes for RPP-WTP and the applicability to systems, structures and components (SSCs) is contained in 24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document (SRD) Volume II, in section 4.1 General Design, Safety Criterion 4.1-3. This Safety Criterion defines Seismic Classes (SC) I, II and III and provides seismic loads and source documents.

Justification: The definition of OBE and SSE are applicable to Nuclear Power Generating Stations and the new definitions are not applicable to the RPP-WTP project as defined in the SRD. This is consistent with the tailoring of AISC N690 as documented in 24590-WTP-ABCN-ESH-01-013. The definition for RIM is retained, as it is required to test active inline devices.

Pages 3-23; Part I, Sections 4-8 and Part II Process and Qualification

Delete all remaining sections of Part I-Process and Part II- Qualification.

Justification: The qualification processes are addressed in IEEE-344-1987(R 1993) and IEEE-323-1983 (R 1990), which are implementing standards of the RPP-WTP project.

Pages 24-27 and 30-32; Sections 1-5 and 7-8

Delete Sections 1-5 and 7-8.

Justification: These sections address tests which do not relate to seismic simulation. These tests are addressed in IEEE-323-1983(R1990) which is an implementing standard of the RPP-WTP project.

Page 27; Section 6.1 Scope

Revise section 6.1 as follows: The seismic simulation test demonstrates the operability of an actuator or in-line mounted instrument during and after exposure to the equivalent dynamic effect of a design basis earthquake.

Justification: The definition of OBE and SSE are applicable to Nuclear Power Generating Stations and the new definitions are not applicable to the RPP-WTP project as defined in the SRD. This is consistent with the tailoring of AISC N690 as documented in 24590-WTP-ABCN-ESH-01-013.

Page 27; Section 6.2 Test setup requirements

Replace a), first sentence as follows: Mount the actuator or inline mounted device to the shake table fixture in the same manner as it would be attached to a valve or mounted in-line.

Justification: IEEE standard 382-1996 provides testing guidance and performance requirements for actuators for power-operated valve assemblies. Current industry practice applies these testing requirements to both actuators and in-line mounted instruments.

Pages 27-28; Sections 6.2 and 6.3 Test setup requirements and Test conduct

Replace all references to "actuator" or "valve actuator" with "valve actuators or in-line mounted instrument".

Justification: IEEE standard 382-1996 provides testing guidance and performance requirements for actuators for power-operated valve assemblies. Current industry practice applies these testing requirements to both actuators and in-line mounted instruments.

Page 28, Section 6.3 Test conduct

Delete paragraphs a), b), and d).

Justification: This test method is used only for line mounted actuators or in-line mounted instruments. Additionally, the definition of OBE and SSE are applicable to Nuclear Power Generating Stations and the new definitions are not applicable to the RPP-WTP project as defined in the SRD. This is consistent with the tailoring of AISC N690 as documented in 24590-WTP-ABCN-ESH-01-013.

Replace reference to "SSE" with "design basis earthquake"

Justification: The definition of OBE and SSE are applicable to Nuclear Power Generating Stations and the new definitions are not applicable to the RPP-WTP project as defined in the SRD. This is consistent with the tailoring of AISC N690 as documented in 24590-WTP-ABCN-ESH-01-013.

Delete all references to "figure 7".

Justification: Figure 7 provides general required response spectra (RRS) to be used when specific RRS for the plant is not available. The RPP-WTP project will generate RRS specific to each facility so the generic RRS provided in figure 7 is not required.

Pages 33-47; Annexes

Delete Annexes A-E.

Justification: These annexes are for informational purposes only.

31.0 IEEE-497, IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations

Revision: 2002
Sponsoring Organization: The Institute of Electrical and Electronics Engineers, Inc.

WTP Specific Tailoring

The following tailoring of IEEE-497 is required for use by the WTP project as an implementing standard for the safety related systems design.

Pages 1-20; All Sections Clarification of Nuclear Power Generating Station Terminology

The terms "Licensing basis documentation (LBD) in the standard apply to the "Authorization Basis (AB) in the WTP.

Justification: As determined by the project contract the LBD on the WTP are classified as "AB".

The term "nuclear plant", "nuclear power generating stations", and "conventional plant" will be taken to mean the WTP.

Justification: Clarifies how the standard will apply to the WTP project.

Page 1; Section 1.1 Scope

Revise section 1.1 as follows:

The criteria increase the specificity of selection requirements, and clarify associated performance and qualification requirements, for accident monitoring instrumentation for WTP.

Justification: IEEE standard 497-2002 provides general selection, performance, design, qualification, display, and quality assurance requirements for accident monitoring instrumentation. The selection requirements need to be restated in terms of WTP terminology and only selection-dependent performance and qualification requirements are retained for clarification. Non-selection-dependent performance criteria, and design, display, and quality assurance criteria are covered with adequate specificity under different sections of SRD and other implementing standards such as ANSI/ISA S84.01-1996, ASME/ANSI standards and IEEE standards.

Page 2; Section 2 References

The following standards are listed in the SRD with revision dates that are different from the revisions dates listed in the standard. The following revisions of the below standards shall be used in place of the revisions referenced in the body IEEE-497.

ASME NQA-1-1989, Quality Assurance Program Requirements for Nuclear Facility Applications

IEEE 308-1991, Criterion for Class 1E Power Systems for Nuclear Power Generating Stations

IEEE 323-1983, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

IEEE 344-1987, Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

IEEE 379-1994, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems

IEEE 384-1992, Standard Criteria for Independence of Class 1E Equipment and Circuits

The following reference standards do not apply for the WTP.

IEEE Std. 603-1998, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations

Justification: IEEE Std. 603-1980 was replaced by ANSI/ISA-S84.01-1996, per 24590-WTP-ABCN-ESH-01-027.

IEEE 7-4.3.2-1993, IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations

Justification: This standard applies for computer systems used in nuclear power generating stations and, due to the rapid advance in computer designs, is out of date for use on the WTP. ANSI/ISA S84.01-1996 will be used on the WTP in place of this standard.

IEEE Std 352-1987 (R1999), IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Stations.

Justification: This standard provides guidelines for Reliability Analysis of Nuclear Power Stations. Standard ANSI/ISA S84.01-1996 provides information more appropriate for WTP on implementing Reliability Analysis and design criteria for Reliability. Therefore, standard ANSI/ISA S84.01-1996 is used on the WTP in place of this standard.

IEEE Std. 577-1976 (R2001), IEEE Standard Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Power Generating Stations.

Justification: This standard provides Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Power Generating Stations. Standard ANSI/ISA S84.01-1996 provides general guidelines and design criteria which are more appropriate for implementing Reliability Analysis on WTP. Therefore, standard ANSI/ISA S84.01-1996 is used on the WTP in place of this standard.

The following reference Standard shall be included:

ANSI/ISA S84.01-1996, Application of Safety Instrumented Systems for Process Industries

Justification: ANSI/ISA S84.01- 1996 replaces IEEE-603 on the WTP, per 24590-WTP-ABCN-ESH-01-027.

Page 5; Section 4.1 Type A variables

Revise sub section (a) as follows:

a) Take specific planned manually-controlled actions for which no automatic control is provided and that are required for SDC and SDS SSCs to perform their safety functions as assumed in the plant AB.

Justification: Type A variables are defined as specific safety related variables that provide the primary information required to permit the control room operating staff to take specific planned actions.

Revise sub section (b) as follows:

b) Not applicable.

Justification: Anticipated operational occurrences (AOOs) are not formally defined in WTP terminology.

Revise last paragraph as follows:

Type A variables provide information essential for the direct accomplishment of specific SDC and SDS safety functions that require manual action. These variables are a subset of those necessary to implement the facility-specific emergency operating procedures (EOPs) or plant abnormal operating procedures (AOPs). Type A variables do not include those variables that are associated with contingency actions that may also be identified in written procedures.

Justification: Type A variables are defined as specific safety related variables that provide the primary information required to permit the control room operating staff to take specific planned actions. The AB, AOPs, and EOPs are the basis for identification of type A variables.

Page 5; Section 4.2 Type B variables

Revise section 4.2 as follows:

Type B variables are those variables that provide primary information to the control room operators to assess the plant SDC and SDS safety functions.

Any plant SDC and SDS functions addressed in the facility-specific emergency operating procedures (EOPs) that are in addition to those identified above shall also be included.

The Type B variables shall be those necessary to implement the facility-specific emergency operating procedures (EOPs) restoration, and the SDC and SDS safety function status trees, if applicable.

Justification: Type B variables are defined as those variables that provide primary information to the control room operators to assess the plant SDC and SDS safety functions. The EOPs are basis for identification of the type B variables.

Page 6; Section 4.3 Type C variables

Revise section 4.3 as follows:

Not Applicable

Justification: This section includes monitoring of three fission product barriers (fuel cladding, reactor coolant system pressure boundary, and containment pressure boundary). The WTP project has no fission product barriers. Containment barriers are covered under variable types A, B or D. Therefore this section is not applicable (N/A).

Page 6; Section 4.4 Type D variables

Revise first paragraph as follows:

Type D variables are SDC, SDS, and RRC related variables that are required in AOPs, EOPs, and the AB to:

Justification: To clarify that this section includes all safety related systems identified in the AB.
Revise Item (b) as follows:

b) Indicate the performance of other systems necessary to achieve and maintain a safe state condition

Justification: To clarify that Type D variables are used for the indication of "safe state" conditions, bringing Standard's wording into conformance with WTP terminology.

Revise second paragraph as follows:

Type D variables shall be based upon the AB and those necessary to implement the following operating procedures:

a) Facility-specific emergency operating procedures (EOPs)
b) Plant AOPs related to AB requirements

Justification: The AB, EOPs, and certain AOPs are the basis for identification of type D variables on WTP. Plant AOPs will be used to implement LCO Action Steps.

Page 6, Section 4.5 Type E variables

Revise section 4.5 as follows:

Not Applicable

Justification: Type E variables are defined as those variables required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases. For WTP, such determination and assessment is covered under other SRD implementing standards.

Page 6, Section 4.6 Documentation of selection bases

Revise section 4.6 as follows:

Documentation shall be developed and maintained for the selection bases of the accident monitoring variables consistent with the plant AB.

Justification: Use WTP terminology.

Page 7, Table 1

Revise Table 1 as follows:

Referenced clause in standard	Selection criteria for the variable type	Source Documents
4.1	Type A – Planned manually controlled actions for accomplishment of SDC and SDS safety functions for which there is no automatic control.	– SDC and SDS safety functions identified in AB – EOPs – AOPs
4.2	Type B – Assess the process of accomplishing or maintaining plant SDC and SDS safety functions	– SDC and SDS safety functions identified in AB – EOPs – AOPs
4.3	Type C Not Applicable	N/A
4.4	Type D – Indicate performance of SDC, SDS, and RRC safety systems – Indicate the performance of required SDC, SDS, and RRC auxiliary support features – Indicate the performance of SDC, SDS, and RRC systems necessary to achieve and maintain a safe state condition – Verify SDC, SDS, RRC safety system status	– SDC, SDS, and RRC safety systems identified in AB – EOPs – AOPs
4.5	Type E Not Applicable	N/A

Justification: Revise to align with changes made in sections 4.1 through 4.5.

Page 8 and 9; Sections 5.1 through 5.3, 5.5 through 5.6, and Annex A Performance Criteria

Revise sections 5.1 through 5.3, 5.5 through 5.6, and Annex A as follows:

Not Applicable

Justification: These sections treat non-selection-dependent performance requirements. The WTP has implemented ANSI/ISA S84.01-1996 in establishing such performance requirements. Therefore, sections 5.1 through 5.3, 5.5 through 5.6, and Annex A of this standard will not be implemented for this project.

Page 9; Section 5.4 Performance Criteria, Required instrumentation duration

Revise sub-section (c) as follows:

Not Applicable

Justification: Revise to align with changes made in sections 4.1 through 4.5.

Revise sub-section (d) as follows:

The post event operating time for Type D variable instrument channels shall be based on the plant's AB.

Justification: Revise to align with changes made in sections 4.1 through 4.5 and WTP terminology.

Page 9; Section 6 Design Criteria

Revise section 6 as follows:

Not Applicable

Justification: This section establishes design requirements for instruments and instrument channels. The WTP has implemented ANSI/ISA S84.01-1996 and the SRD in establishing these design criteria. Therefore, section 6 (design criteria) of this standard will not be implemented for this project.

Page 14; Section 7 Qualification Criteria

Revise first paragraph as follows:

The requirements for equipment qualification (seismic and environmental qualification) of accident monitoring instruments shall be consistent with their AB-, EOP- or AOP-based monitoring function during and following a design basis event (including seismic events). Such requirements shall be in addition to any qualification requirements otherwise applicable as a result of the instruments' safety function and classification as SDC, SDS, or RRC.

Justification: Provide a basis for upgrading instrument qualification requirements based on accident monitoring requirements, beyond the normal safety function and SDC, SDS, or RRC basis. Bring into alignment with WTP terminology.

Revise sections 7.1 through 7.8 as follows:

Not applicable

Justification: These sections establish requirements for equipment seismic and environmental qualification covered elsewhere in the SRD. Therefore, these sections will not be implemented for this project.

Page 15; Section 8 Display Criteria

Not Applicable

Justification: This section establishes display requirements covered elsewhere in the SRD. Therefore, section 8 (display criteria) of this standard will not be implemented for this project.

Page 18; Section 9 Quality assurance

Not Applicable

Justification: This section establishes quality assurance requirements. The WTP project follows requirements establish in SRD. Therefore, section 9 (quality assurance) of this standard will not be implemented for this project.

32.0 ISO American Petroleum Institute Standards*

32.1 API Standard 610, Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services, Eighth Edition, August 1995.

WTP Specific Tailoring

The following tailoring of API 610 is required for use as an implementing standard for centrifugal pumps handling radioactive waste streams required to Safety Criterion 4.2-2 confinement requirements.

API 610 Sections 2.2 Pressure Casings, 2.4 External Nozzle Forces and Moments, 2.7 Mechanical Seals, 2.11 Materials, 3.5 Piping and Appurtenances, 4.2.2 Material Inspection, and 4.3.2 Hydrostatic Test

Centrifugal pumps which provide a confinement function in accordance with Safety Criterion 4.2-2 shall meet the requirements of sections 2.2, 2.4, 2.7, 2.11, 3.5, 4.2.2, and 4.3.2 of API Standard 610-1995, Eighth Edition.

Justification: The API sections listed above are required to meet the SRD 4.2-2 requirements for confinement design for Important to Safety centrifugal pumps. This approach ensures that pumping equipment supplied and installed in the WTP can be relied upon to maintain confinement of radioactive process streams during operating conditions including shutdown. ASME Sections II, V, VIII, and IX are referenced in these standards as the acceptance standards for the materials, design, welding, heat treating, and inspection.

API Standard 610, section 2.2 pressure casings, referencing ASME Section VIII, Div. 1, requires that stress used in the design for any given material shall not exceed ASME Section II values for the same material.

API Standard 610, section 2.4, provides the allowable nozzle loadings.

API Standard 610, sections 2.7 requires that mechanical seals shall be furnished unless otherwise specified; and unless otherwise specified, seals and sealing systems to be furnished in accordance with API Standard 682; and when they do not comply with API Standard 682, seals shall meet the requirements of API Standard 610 sections 2.7.3.1 through 2.7.3.23.

API Standard 610, section 2.11, specifies ASME Section VIII, Div. 1, and ASME Section IX, for the materials, casting factors, welding and weld quality, and low temperature requirements and provides the acceptance standards for inspecting the pressure boundry of WTP pumps.

API Standard 610, section 3.5, specifies that the piping design, materials, joint fabrication, examination, and inspection be in accordance with ASME B31.3.

API Standard 610, section 4.2.2 states that material inspection including radiography, ultrasonic, magnetic particle, and liquid penetration inspection shall meet the acceptance standards of ASME Section VIII and/or ASME Section V.

The API Standard 610, section 4.3.2 requires the hydrostatic test to be maintained for a minimum of 30 minutes at 1.5 times maximum allowable working pressure for leaks or seepage through the casing or casing joint, and it is more stringent than ASME Section VIII or ASME B31.3-1996 for pressure boundary testing. API Standard 610, section 4.3.2 references ANSI/ASME B31.3 or ASME Section II, Div.I for arriving at the material properties used test pressures.

API Standard 610 sections 2.2, 2.4, 2.7, 2.11, 3.5, 4.2.2, and 4.3.2 include the applicable ASME Section VIII, Div. 1, and ASME B31.3, requirements, and provide adequate requirements to ensure the confinement design for Important to Safety centrifugal pumps.

32.2 API Standard 685, Sealless Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services, First Edition, October 2000.

WTP Specific Tailoring

The following tailoring of API 685 is required for use as an implementing standard for centrifugal pumps handling radioactive waste streams required to Safety Criterion 4.2-2 confinement requirements.

Sections 6.3 Pressure Casings, 6.5 External Nozzles Forces and Moments, 6.11 Materials, 6.12 Castings, 6.13 Welding, 6.14 Low Temperature, 7.3 Piping and Appurtenances, 8.2.2 Material Inspection, and 8.3.2 Hydrostatic Test

Sealless centrifugal pumps which provide a confinement function in accordance with Safety Criterion 4.2-2 shall meet the requirements of sections 6.3, 6.5, 6.11, 6.12, 6.13, 6.14, 7.3, 8.2.2, and 8.3.2 of API Standard 685-2000.

Justification: The API sections listed above are required to meet the SRD 4.2-2 requirements for confinement design for Important to Safety sealless pumps. This approach ensures that pumping equipment supplied and installed in the WTP can be relied upon to maintain confinement of radioactive process streams during operating conditions including shutdown. ASME Sections II, V, VIII, and IX are referenced in these standards as the acceptance standards for the materials, design, welding, heat treating, and inspection.

API Standard 685, section 6.5, provides the allowable nozzle loadings.

API Standard 685, section 6.3 pressure casings, referencing ASME Section VIII, Div. I, requires that stress used in the design for any given material shall not exceed ASME Section II values for the same material.

API Standard 685, section 6.11 specifies ASME Section VIII, Div. 1 and ASME Section IX, for the materials, casting factors, welding, and weld quality to be used as the acceptance standards for maintaining pressure integrity of WTP pumps.

API Standard 685, section 6.12 allows the repair of weldable steel used in castings provided it is done in accordance with ASME Section IX.

API Standard 685, section 6.13 requires that welding of piping, pressure-containing parts, and wetted parts and heat treatment of welds shall be performed in accordance with ASME Section VIII and ASME Section IX.

API Standard 685, section 6.14 requires that pumps operated at a low temperature comply with the material requirement in ASME Section VIII.

API Standard 685, section 7.3 specifies that the piping design, materials, joint fabrication, examination, and inspection be done in accordance with ASME B31.3.

API Standard 685, section 8.2.2 states that for material inspection including radiography, ultrasonic , magnetic particle, and liquid penetration inspection shall meet the acceptance standard used for casting per ASME Section V or ASME Section VIII, Div. 1.

API Standard 685, section 8.3.2 requires the hydrostatic test to be maintained for a minimum of 30 minutes at 1.5 times maximum allowable working pressure for leaks or seepage through the casing or casing joint, and it is more stringent than ASME Section VIII or ASME B31.3-1996 for pressure boundary testing. Section 8.3.2 references ANSI/ASME B31.3 or Section II, Division 1 of the ASME Code for arriving at test pressures.

API Standard 685 sections 6.3, 6.5, 6.11, 6.12, 6.13, 6.14, 7.3, 8.2.2, and 8.3.2 include the applicable ASME Section VIII, Div.1, and ASME B31.3, requirements, and provide adequate requirements to ensure the confinement design for Important to Safety sealless pumps.

Appendix D

Radiological Exposure Standards for the WTP Project

CONTENTS

1.0 Introduction and Purpose

This attachment to the SRD originally was issued as a stand-alone document (BNFL-5193-RES-01, Revision 0, dated 28 August 1997). It has been incorporated into the SRD because it provides both background information and the basis for the radiological exposure standards reflected in the SRD Safety Criteria. In addition, it has been updated to reflect responses to DOE questions on the Standards Approval Package. It has also been updated to reflect a change in the radiological exposure standards for facility workers in the extremely unlikely event frequency range.

This document is the Radiation Exposure Standard for Workers under Accident Conditions, which is a radiological safety deliverable. This document is used during the process hazards analysis (PHA) and accident analysis to ensure worker safety through identification of the need for accident prevention and mitigation features that provide worker protection against radiological and nuclear hazards. In this document, where unmodified reference is made to workers, it applies collectively to facility workers and collocated workers as defined in sections 3.5.1 and 3.5.2 below.

The US Department of Energy (DOE), in DOE/RL-96-0006, Revision 0, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors*, (DOE-RL 1996), provides Table 1, "Dose Standards Above Normal Background". In Table 1 (referred to as DOE Table 1), there are entries labeled, "To be derived", for which the contractor is to propose specific exposure standards for both facility workers and collocated workers for the following events:

- **Unlikely Events**: events that are not expected but may occur during the lifetime of the facility in the range of frequency between 10^{-2}/yr and 10^{-4}/yr (between once in 100 years and once in 10,000 years)
- **Extremely Unlikely Events**: events that are not expected to occur during the lifetime of the facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Extremely unlikely events are in the range of frequency between 10^{-4}/yr and 10^{-6}/yr (between once in 10,000 years and once in 1 million years).

This document provides the required exposure standards and the bases for their selection. In addition, this document presents the approach for complying with DOE Table 1. The individual elements of this approach, as shown in Table 2-1 of SRD Safety Criterion 2.0-1 (referred to as Table 2-1), are conservative based on the requirements of the contract and, as such, satisfy the contract. For completeness, this document also discusses, and presents in Table 2-1, public exposure standards and the assumed locations of the public, facility worker, and collocated worker for use in evaluation of accident consequences and normal radioactive material releases.

2.0 Exposure Standards for Facility and Collocated Workers

The four "To be derived" cells in DOE Table 1 have been completed by imposing a radiological exposure standard not to exceed 25 rem/event to the WTP facility workers or to collocated workers for unlikely events, 100 rem/event to the WTP facility workers for extremely unlikely events, and 25 rem/event to the WTP co-located workers for extremely unlikely events.

The 25 rem/event exposure standard for both the facility and collocated workers for unlikely events corresponds to the once-in-a-lifetime accident or emergency exposure for radiation workers which, by recommendation of the National Committee on Radiation Protection (NCRP 1963), may be disregarded in the determination of their radiation exposure status. In addition, an exposure of 25 rem/event corresponds to a conditional probability of fatality of about 2×10^{-2}. For unlikely events (defined in Table 2-1 as having a maximum occurrence frequency of 10^{-2}/yr), this equates to a maximum increase in worker lifetime risk of premature death of only 2×10^{-4}, which is considerably less than the average accidental death risk for workers in some of the safest industries (i.e., retail and wholesale trade, manufacturing, and service [EPA 1991]).

The 100 rem/event exposure standard for the facility workers for extremely unlikely events is consistent with the worker exposure standard being employed elsewhere in the DOE complex including the Hanford Site. In addition, an acute radiation dose of approximately 100 rem carries almost no risk of prompt death [DOE 1994a].

Compliance with these worker exposure standardsare established using qualitative methods supported, where necessary, by numerical analysis that may include the development of event trees and fault trees and/or the performance of consequence analyses. From this process, preventative and mitigative engineered and administrative controls are identified.

Use of qualitative methods is consistent with the American Institute of Chemical Engineers (AIChE) guidelines (AIChE 1992), US Nuclear Regulatory Commission (NRC) guidance for the performance of integrated safety analysis for 10 *Code of Federal Regulations* (CFR) 70 special nuclear material licensees (NRC 1995a), as well as DOE-STD-3009 (DOE 1994) and DOE G 420.1-X (DOE 1995). Both DOE documents state the following:

> "Estimates of worker consequences for the purpose of a safety-significant SSC designation are not intended to require detailed analytical modeling. Considerations should be based on engineering judgement of possible effects and the potential added value of safety-significant SSC designation."

Because the primary purpose of the WTP Project facility and collocated worker exposure standards is to identify structures, systems, and components (SSC) required to protect these workers, the guidance cited above is both applicable and appropriate.

The principal approach for complying with the worker exposure standard is the PHA. The PHA is a systematic, team-based review of the plant and treatment processes. The PHA identifies hazards and operability problems to a level of detail commensurate with the design detail available. Further hazard evaluation takes place in parallel with design development to ensure that safety continues to be built into the design process.

Having generated the list of hazards and hazardous situations, this list is subject to a further systematic team-based review where a binning process takes place. The binning process assigns postulated events to a certain severity level for further detailed analysis and comparison to radiation exposure standards.

The worker exposure standards for unlikely or extremely unlikely events apply to events with frequencies less than 10^{-2}/yr. For those frequencies, the PHA process assigns serious and major hazardous situations as undesirable, acceptable with controls, or acceptable. For a hazardous situation to be "acceptable", its consequences must be less than the corresponding worker exposure standard. Where there is uncertainty as to where an event should be binned (i.e., assigning a hazard category), it is binned into a higher category to ensure that the accident analysis remains conservative.

The DOE-RU has provided a guidance document (DOE-RL 1997) to be used for review of the Radiation Exposure Standard for Workers Under Accident Conditions. This guidance document includes the accident risk goal of DOE/RL-96-0006.

DOE/RL-97-09 (DOE-RL 1997) describes approaches that can be taken to meet this goal. The simplest approach notes that the goal can be met when (a) a worker dose standard that does not exceed 100 rem is used for extremely unlikely events (10^{-4} to 10^{-6} probability range), and (b) a worker dose standard that does not exceed 10 rem is used for unlikely events (10^{-2} to 10^{-4} probability range). For the latter probability range, the 10-rem standard relies on the assumption that the probability of accidents is evenly distributed across the probability range.

Based on experience with similar plants, it is considered unlikely that the even distribution assumption will represent the actual situation for WTP. Furthermore, experience indicates that there will be relatively few accidents falling into this range, and that they will be distributed toward the low probability end of the range. Consequently, a value higher than 10 rem can be used for the worker accident standard for unlikely events.

As can be seen in Table 2-1, a value of 25 rem/event is selected as the worker accident standard for unlikely events.

The accident risk goal is stated in DOE/RL-96-0006 as, "The risk, to an average individual in the vicinity of the Contractor's facility, of prompt fatalities that might result from an accident should not exceed one-tenth of one percent (0.1 %) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed." The DOE guidance document states that a radiation exposure standard of 100 rem/event would satisfy the accident risk goal. Because the WTP standard is 100 rem/event, the guidance document is satisfied.

In each of the four cells addressing accident exposure standards for workers and collocated workers in the unlikely and extremely unlikely events ranges, an ALARA accident limit is not specified. However, Note 2 of Table 2-1 states:

> "In addition to meeting the listed dose standards for accidents, the approach to accident mitigation is to evaluate accident consequences to ensure that the calculated exposures are far enough below standards to account for uncertainties in the analysis, and to provide for sufficient design margin and operational flexibility."

This approach provides an adequate level of safety. The following paragraphs should also be noted in support of this conclusion.

The accident analyses will show compliance with exposure standards for accidents. In addition, a defense-in-depth approach provides multiple levels of protection that ensure worker exposures from accidents will be significantly lower than calculated. This is a proven approach, considered to be effective at minimizing exposures to workers.

The approach to accident mitigation (as described in Note 2 of Table 2-1) is to examine accident consequences to ensure that calculated exposures are far enough below standards to account for uncertainties in the analysis and to provide sufficient design margin and operational flexibility. This approach is employed for all accidents (including both public and workers at all accident frequency levels) that can challenge the exposure standards, ensuring that accident exposures would be well below standards.

3.0 Development of the BNI Approach to Compliance with Table 1 of DOE/RL-96-0006

The overall approach to complying with DOE Table 1 is presented in this document. This approach takes the form of Table 2-1. The "To be derived" cells have been completed as discussed. The remaining cells of Table 2-1 are either identical or conservative with respect to DOE Table 1. The following sections discuss differences between DOE Table 1 and Table 2-1.

DOE Table 1 footnotes are not shown in Table 2-1. Section 2.1 of DOE/RL-96-0006 states that the *footnotes refer only to the origin of the specific* standards and, as such, are not considered contractual requirements unless included elsewhere in the contract.

3.1 Estimated Frequency of Occurrence

The second column of DOE Table 1, "Estimated Probability of Occurrence (P) (yr^{-1})," has been titled in Table 2-1, "Estimated Frequency of Occurrence (f) (yr^{-1})". In addition, the estimated frequency of occurrence for normal events of DOE Table 1 is redefined in Table 2-1 as any normal event regardless of frequency (nominally taken to be a frequency > 0.1/yr). The estimated frequency of anticipated events in DOE Table 1 is redefined as events with an annual frequency of occurrence of $10^{-2} < f\ 10^{-1}$.

With these changes, events routinely performed (e.g., melter replacement) are considered normal events rather than accidents, irrespective of frequency of occurrence. As normal events, the radiological assessment is subject to the more restrictive "per year" exposure standards rather than "per event" exposure standards. Consequently, these changes are conservative in comparison to DOE Table 1.

3.2 Normal Events/Public and Workers Exposure Standards

Clarifying descriptions have been included in the Normal Events/Public cell of Table 2-1 explaining that the second 100 mrem/yr standard applies to a member of the public entering the controlled area and the 25 mrem/yr standard is the public primary exposure standard for radioactive waste. The removal of DOE Table 1 footnotes (as noted above) necessitated the addition of these clarifying notes.

For the Normal Events/Worker and Normal Events/Collocated Worker cells of Table 2-1, the DOE Table 1 standard of 1.0 rem/yr ALARA design limit is replaced by a standard of 1.0 rem/yr ALARA design objective per 10 CFR 835, section 1002(b). The corresponding worker standards for normal events in DOE Table 1 are tied to the ALARA design objectives of 10 CFR 835.1002(b) by the footnotes to DOE Table 1.

BNI has committed to full compliance with 10 CFR 835 in the SRD, and the other sections of 10 CFR 835.1002 provide adequate requirements to ensure routine worker exposures will be ALARA. In addition, a footnote, Note 1, is included in Table 2-1. This note states the following:

> "In addition to meeting the listed design objective of 10 CFR 835.1002(b), the inhalation of radioactive material by workers and collocated workers under normal conditions is kept ALARA through the control of airborne radioactivity as described in 10 CFR 835.1002(c)."

3.3 Anticipated Events/Worker and Collocated Worker Exposure Standards

References to as low as reasonably achievable (ALARA) standards have been removed for the Anticipated Events/Worker and Collocated Worker cells of Table 2-1. The ALARA design objective of 10 CFR 835, "Occupational Radiation Protection", is applied to normal events as shown in Table 2-1. However, with the redefinition in Table 2-1 of anticipated events as those events with an annual frequency of occurrence of $10^{-2} < f \leq 10^{-1}$, the ALARA objective no longer applies because anticipated events are not part of normal operation.

This change complies fully with section 3.2, "Radiation Protection Objective", of DOE/RL-96-0006, which states the following:

> "Ensure that during normal operation radiation exposure within the facility and radiation exposure and environmental impact due to any release of radioactive material from the facility is kept as low as is reasonably achievable (ALARA) and within prescribed limits, and ensure mitigation of the extent of radiation exposure and environmental impact due to accidents."

This aspect of Table 2-1 also represents compliance with contractual requirements because footnote 3 of DOE Table 1 references 10 CFR 835.1002(b). This section, and 10 CFR 835.202 which it references, establishes design requirements for occupational exposures other than planned special exposures and emergency exposures. Administrative limits for planned special exposures and emergency exposures are addressed in 10 CFR 835.204 and 10 CFR 835.1302 and are complied with by the WTP.

Finally, to provide an adequate level of safety and to ensure that cost-effective safeguards affecting anticipated events are evaluated (and incorporated as appropriate) whenever the final calculated event consequence to a worker or collocated worker is 1 rem or more, the approach specifies a 1.0-rem/event design action threshold standard. In addition, a note is included in Table 2-1 to explain the application of the standard. This note (Note 3 to Table 2-1) states:

> "When a calculated accident exposure exceeds this threshold, then appropriate actions are taken. These include carrying out a less bounding (i.e., more realistic) evaluation to show that the accident consequences will be below the threshold or evaluating additional safeguards for cost-effectiveness and/or feasibility. This threshold is not a limit; it does not require the implementation of additional preventative or mitigative features if they are not both cost-effective and feasible."

3.4 Extremely Unlikely Events/Public Exposure Standard

A standard is included in the Extremely Unlikely Events/Public cell of Table 2-1 stating that a public exposure standard target value of 5 rem/event is applied to extremely unlikely events. This target value is based on the following:

- The philosophy is that the public should be protected by a lower exposure standard than a worker. This philosophy recognizes the fact that the worker has agreed to work on the Hanford Site and has received training for avoiding hazards and dealing with hazardous situations.
- A goal to facilitate transition to the NRC as the regulatory agency with jurisdiction over nuclear safety for DOE facilities. With the exception of a 25 rem/event guideline value of 10 CFR 100 for the establishment of the exclusion area and low population zone for commercial power reactors, the NRC has not established a public exposure standard that exceeds 5 rem/event. A public exposure standard of 5 rem/event is also included in proposed rulemaking for 10 CFR 70 (NRC 1995b), which further supports the Table 2-1 value.
- With the same 5 rem/event public exposure standard for both unlikely and extremely unlikely events, there is no need to bin accidents in one of these two event frequency categories for the purpose of establishing protection of public safety.

3.5 Location of Receptors

In Table 2-1, a new last row has been added to clarify in DOE Table 1 of DOE/RL-96-0006 the assumed location for the facility worker, the collocated worker, and the public, for the purpose of establishing compliance with the radiological standards of DOE Table 1. The bases for the receptor locations included in this row are provided below.

3.5.1 Facility Worker

The facility worker is located at the most limiting location within the WTP contractor-controlled area as defined in DOE/RL-96-0006, as shown in SRD Safety Criterion 2.0-1, Figure 1.

Figure D-1 Deleted (Moved to SC 2.0-1)

Section 6.0, "Glossary", of DOE/RL-96-0006 defines the controlled area as the following:

> "The physical area enclosing the facility by a common perimeter (security fence). Access to this area can be controlled by the Contractor. The controlled area may include identified restricted areas."

The controlled area for WTP used to define the location of the facility worker, is that land within the WTP security fence.

3.5.2 Collocated Worker

Section 6.0, "Glossary", of DOE/RL-96-0006 defines the collocated worker as the following:

> "An individual within the Hanford Site, beyond the Contractor-controlled area, performing work for or in conjunction with DOE or utilizing other Hanford Site facilities."

For evaluation of the WTP design to the exposure standards of DOE Table 1, the location of the collocated worker is either at the controlled area boundary or beyond that boundary if such a location results in higher exposure. For a ground-level release, the location of the collocated worker is considered no closer than 100 m from the release point.

3.5.3 Public

The location of the public (i.e., the offsite receptor) for the purpose of establishing compliance with the last column of DOE Table 1 of DOE/RL-96-0006, is established at the most limiting exposure location along the near bank of the Columbia River, Highway 240, and a southern boundary as shown in SRD Safety Criterion 2.0-1, Figure 2.

This area includes land for which it is reasonable to assume DOE will retain the right to control activities and limit access under accident conditions for the operating life of the WTP. Specifying the near river bank excludes the Columbia River for which DOE does not control activities (DOE-RL 1995). Specifying Highway 240 excludes the Arid Lands Ecology Reserve of which DOE might relinquish control during the operating life of the WTP. The southern boundary serves to exclude Energy Northwest's Columbia Generating Station, a commercial nuclear power plant (whose workers should be considered members of the public), and the Hanford Site 300, 400, and 1100 Areas. The 400 Area includes the Fast-Flux Test Facility.

Figure D-2 Deleted (Moved to SC 2.0-1)

In footnotes 10 and 12, DOE Table 1 of DOE/RL-96-0006 makes reference to 10 CFR 72, "Licensing Requirements for the Independent Spent Fuel (ISFSI) and High Level Radioactive Waste," and 10 CFR 100, "Reactor Site Criteria," to relate to the public exposure standards for unlikely and extremely unlikely events. While the siting requirements and guidance of Parts 72 and 100 are not applicable to the WTP, the requirements for establishing the location of the offsite receptor in these two cited regulations are useful for locating the offsite receptor for a waste processing facility such as WTP. Section 72.106, "Controlled Area Boundary of an ISFSI or Monitored Retrievable Storage (MRS)", includes the following statements relative to the boundary to be assumed for the evaluation of radiological exposure to the public:

> "The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters."
>
> "The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety."

Title 10 CFR 100 establishes a guideline value of 25 rem for 2 hr at the exclusion area boundary. For the exclusion area, 10 CFR 100.3, "Definitions", states the following:

> "(a) *Exclusion area* means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result."

As can be seen from the above excerpts, the assumed location for the offsite receptor for WTP is consistent with 10 CFR 72 and 10 CFR 100. In addition, the proposed southern boundary takes advantage of the road junction at the Wye barricade SRD Safety Criterion 2.0-1, Figure 2 for control of access to the site during accident conditions.

4.0 References

10 CFR 70, "Domestic Licensing of Special Nuclear Material", *Code of Federal Regulations*, as amended.

10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste", *Code of Federal Regulations*, as amended.

10 CFR 100, "Reactor Site Criteria", *Code of Federal Regulations*, as amended.

10 CFR 835, "Subpart C - Standards for Internal and External Exposure", *Code of Federal Regulations*, as amended.

AIChE, 1992, *Guidelines for Hazards Evaluation Procedures, Second Edition with Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, New York.

DOE 1994, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, DOE-STD-3009-94, US Department of Energy, Washington, DC.

DOE 1994a, *Methods for the Assessment of Worker Safety under Radiological Accident Conditions at Department of Energy Nuclear Facilities*, EH-12-94-01, US Department of Energy, Office of Environment, Safety and Health, Office of Nuclear Safety, Washington, DC.

DOE 1995, *Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria*, DOE G 420.1-X, Revision G, US Department of Energy, Washington, DC.

DOE-RL 1995, *Clarification of Hanford Site Boundaries for Current and Future Use in Safety Analysis*, letter Walter B. Scott, DOE-RL to Contractors, dated 26 September 1995, US Department of Energy, Richland Operations Office, Richland, Washington.

DOE-RL 1996, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors*, DOE/RL-96-0006, Revision 0, US Department of Energy, Richland Operations Office, Richland Washington.

DOE-RL 1997, *Guidance for Review of TWRS Privatization Contractor Radiation Exposure Standards for Workers*, DOE/RL-97-09, US Department of Energy, Richland Operations Office, Richland Washington.

EPA 1991, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, US Environmental Protection Agency, Washington, DC.

NCRP 1963, *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure*, Handbook 69, Addendum 1, National Bureau of Standards, Washington, DC.

NRC 1995a, *Integrated Safety Analysis Guidance Document*, NUREG-1513, Draft, US Nuclear Regulatory Commission, Washington, DC.

NRC 1995b, *Preliminary Working Draft of Revision of 10 CFR 70 Updated*, 5 April 1995, provided at the NRC public meeting of May 2, 1995, US Nuclear Regulatory Commission, Washington, DC.

Appendix E

Reliability, Availability, Maintainability, and Inspectability (RAMI)

To ensure that the facility meets operational requirements, it is necessary to address issues associated with reliability, availability, maintainability, and inspectability.

Reliability is used as a measure of the ability of an item or system to complete a task, and it is normally expressed as a probability of failure. Reliability is designed in through the use of appropriate design techniques and control of the mode of operation and the environment. Design techniques to be used vary because they are dependent on the specific item or system and the task to be performed. Their purpose is to optimize reliability by the following:

1) Use of proven materials and components
2) Design simplicity
3) Testability
4) Control of manufacturing standards
5) Control of operational mode (e.g., prevention of misuse and overloads)
6) Control of environment (e.g., protection against corrosion and vibration)

Consistent with the WTP process for tailoring hazard controls using the potential radiological and chemical consequences of individual events, reliability is assigned to SSCs based upon the importance of the SSC to the prevention or mitigation of accidents. The significance of accident prevention and mitigation is determined by the severity of the accident to workers or the public. To implement this tailoring in a clear, consistent, and defensible manner, an Implementing Standard for Safety Standards and Requirements Identification was developed. This Implementing Standard includes a Severity Level ranking system which provides the hazard assessment and control teams with a defined way to categorize the potential severity of those events that can result in radiological or hazardous exposure to the workers or the public. The Implementing Standard provides the means by which the hazard assessment and control teams establish target reliabilities for SSCs.

Availability is a measure of the degree to which an item or system is in an operable condition. It is expressed quantitatively as the ratio of the mean time between failures to the sum of the mean time between failures and the mean time to repair. System availability is calculated to determine the potential for downtime. In this way, systems are identified that contribute to decreased availability. Required availability is achieved by specifying additional systems or increasing reliability of existing systems.

Maintainability is the relative ease and economy of time and resources with which an item can be retained in, or restored to, a specified condition when maintenance is performed by personnel having specified skill levels, using prescribed procedures and resources, at each prescribed level of maintenance and repair. In this context, it is a function of design. Although other factors, such as highly trained people and a responsive supply system, can help keep downtime to an absolute minimum, it is the inherent maintainability that determines this minimum. Improving training or support cannot effectively compensate for the effect on availability of a poorly designed (in terms of maintainability) product. Minimizing the cost to support a product and maximizing the availability of that product are best done by designing the product to be reliable and maintainable.

Inspectability is the measure of the ease with which items or systems can be inspected for preventative maintenance or assessment of condition. Inspectability is used to monitor facility items in order to maintain their reliability. Inspectability of facility items can be designed in by the use of shielded access areas (as above, to reduce radiation exposure) for active equipment or the provision of monitoring equipment (e.g., material coupons for determining vessel corrosion rates, and in-cell cameras).

During the design phase, the WTP facility and processes are evaluated for reliability, availability, maintainability, and inspectability. A number of validated modeling techniques (computer codes, mathematical modeling, failure modes, and effects analysis) for determining reliability and availability of the facility and processes are used. These are used to identify those facility and process areas that are sensitive with respect to influencing overall facility and process performance. Optimum reliability is established by the use of appropriate standards and quality control. The determination of maintenance and inspection needs is based on facility and process reliability requirements. It is a mixture of process optimization, provision of appropriate design features to aid preventative and scheduled maintenance and inspection, and the development of maintenance and inspection programs (administrative and procedural controls) whose objectives among other things, are to facilitate these activities. Reliability targets are assigned to SSCs only when a quantitative value has been credited for the reliability of an SSC in safety analysis.

Appendix F

Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning

CONTENTS

1.0 Introduction

All elements of the WTP safety approach are applied to the deactivation phase of the project. In addition, the WTP will incorporate design provisions to facilitate deactivation and final decommissioning as described in the implementing standard DOE G 441.1-2, *Occupational ALARA Program Guide*, for SRD Criterion 8.0 - 2. These provisions will reduce radiation exposure to Hanford Site personnel and the public during and following deactivation and decommissioning activities and minimize the quantity of radioactive waste generated during deactivation. The purpose of this standard is to define the attributes that must be addressed during the preparation of the deactivation plan to protect both the Hanford Site personnel and the public both during and after the deactivation stage of the project.

2.0 Plan Preparation

A deactivation plan will be prepared prior to construction of the WTP. The deactivation plan will provide details on how the following activities will be accomplished to achieve a deactivated status for the facility.

1) Verification of the completion of the facility deactivation end point. The term facility deactivation end point refers to the set of conditions that comprise the completion of facility deactivation i.e., radiological, structural, equipment, and documentation. These general end points will be defined in the deactivation plan and a requirement made to determine specific end points. When these end point criteria are met the facility will be in a safe state that can be economically monitored and maintained until final decommissioning.
2) Documentation of the regulatory status, conditions, and inventories of remaining radioactive and hazardous materials and health and safety requirements. After facility construction but before deactivation commences, the deactivation plan will require a hazard evaluation for radiological, nuclear, and process safety be carried out. Safety standards and requirements will be identified to implement the controls to protect against the facility hazards.
3) Identification of the facilities, structures, support systems, and surveillance systems to provide for confinement and monitoring of the remaining contamination, radiation, and other potential hazards. After facility construction but before deactivation commences, the plan will be expanded to describe the activities required to maintain the operability of critical equipment and to maintain the structural integrity of the deactivated facility. It will identify modification requirements to systems for the above purposes.
4) Posting and securing of the facility. After facility construction but before deactivation commences, the plan will be expanded to identify the radiological controls required for the deactivated facility, which will include posting of radiological areas. The need for other safety postings will also be identified.
5) Removal of packaged special nuclear materials and other packaged radiological and chemical materials.
6) Confirmation that security systems and procedures are adequate and in place to prevent unauthorized entry.

7) Waste minimization during the deactivation process.

3.0 Summary

The above requirements for the deactivation plan in combination with measures taken at the design stage of the project will protect the Hanford Site personnel and the public both during and following the deactivation activities.

4.0 Definitions

Deactivation - Placing the facility in stable and known conditions, identifying hazards, eliminating or mitigating hazards, and transferring programmatic and financial responsibilities from the operating program to the disposition program. Surveillance and maintenance continues to assure public, environment, and worker safety. The facility is in a safe storage mode, with ongoing, low levels of surveillance and maintenance. The general intent is that the facility be unoccupied and locked except for periodic inspections. Radioactive and hazardous materials may remain in the facility and are subject to ongoing regulatory oversight. (DOE/EM-0318, *Facility Deactivation Guide -- Methods and Practices Handbook*, December 1996)

Decommissioning - The process of removing a facility from operation, followed by decontamination, entombment, dismantlement, or conversion to another use. (DOE G 430.1-1A, *Life Cycle Asset Management*)

Decontamination - The reduction or removal of contaminating radioactive material from a structure, area, object or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination, (2) letting the material stand so that the radioactivity is decreased as a result of natural decay, and (3) covering the contamination to shield or attenuate the radiation emitted. (Health Physics and Radiation Health Handbook, Revised Edition, Bernard Shleien, 1992)

End Point - Specifying and achieving end points is a systematic, engineering way of proceeding from an existing condition to a stated desired final set of conditions in which the facility is safe and can be economically monitored and maintained. (DOE/EM-0318, *Facility Deactivation Guide - Methods and Practices Handbook*, December 1996)

Appendix G

Ad Hoc Implementing Standard for Safety Analysis Reports

CONTENTS

1.0 Introduction

The purpose of this Implementing Standard is to define the format and content for WTP safety analysis reports (SARs).

Section 2.0 provides the definitions important to this Implementing Standard. Section 3.0 defines the process for development, review, and approval.

2.0 Definitions

For the definitions of the following terms, see the reference provided.

Safety Analysis Report (SAR) (DOE/RL-96-0006 [DOE-RL 1998b])

3.0 Process

3.1 Safety Analysis Report Preparation

The River Protection Project Waste Treatment Plant (WTP) SARs document the safety analyses for the facility to demonstrate that it can be safety operated, maintained, and shut down.

The SARs shall be prepared in accordance with the requirements of:

1) DOE/RL-96-0003, *DOE Regulatory Process for Radiological, Nuclear, and Process Safety for TWRS Privatization Contractors* (DOE-RL 1998a), sections 4.3.2 and 4.3.3, both titled "Contractor Input"
2) Contract Table S7-1, "Radiological, Nuclear, and Process Safety Deliverables"
3) *Safety Requirements Document Volume II* (SRD) (BNI 2001), Safety Criterion 9.1-2

The content of the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR) are developed using the guidance provided in the Nuclear Regulatory Commission's 1995 draft revision to Regulatory Guide 3.52, *Standard Format and Content for Heatlh and Safety Sections of License Applications for Fuel Cycle Facilities* (NRC 1995). The content of the SARs is tailored to the nature of the WTP relative to the hazards and hazardous situations identified by the process hazards analysis. Planned deviations from the content guidance of draft Regulatory Guide 3.52 are identified in Table G-1.

The Table of Contents for the safety analysis reports follows Table G-1. The safety analysis report will not be submitted to the regulator until all major safety issues have been resolved and other safety issues have been scheduled for completion. The FSAR should identify significant changes made in the facility design and plans for operation from what was presented in the PSAR. The FSAR, in addition to including facility and process drawings, should also include fabrication and construction specifications important to the safety analysis of the facility.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
1.3 Site Description	Regulatory Guide (RG 3.52) suggests that section 1.3 summarize information used in preparing the Environmental Report. Specific information is referenced, but not duplicated in the safety analysis report (SAR).	The Environmental Report provides this information.
1.3.2 Demography and Land Use	The population distribution as a function of distance and direction is not to be provided. The distances to nearby population centers are provided.	There are no residences on the Hanford Site and the nearby population is low.
3.3 Quality Assurance	Section 3.3.4, "Quality Assurance Program Description" addresses the 10 criteria of 10 CFR 830 Subpart A, "Quality Assurance Requirements" in lieu of the 18 criteria listed in RG 3.52.	By contract compliance to the 10 CFR 800 series of nuclear safety requirements is required. This includes compliance to 10 CFR 830 Subpart A, "Quality Assurance Requirements". The differences in the criteria to be addressed are not significant because the quality assurance programs are based on consensus standards.
3.5 Human Factors	RG 3.52 states that a formal human factors program is not required if the facility has no requirement for safety-class actions. Human factors are considered in the Preliminary Safety Analysis Report (PSAR) independent of whether or not human actions are required for protection of the public or workers.	The requirements of DOE/RL-96-0006 (DOE-RL 1998a), section 4.2.6, "Human Factors", extend beyond consideration of human factors as related to actions taken to protect the public. Final Safety Analysis Report (FSAR) section 3.5 documents how compliance to contract section 4.2.6 is achieved.
3.10 Testing Program and Preoperational Safety Review	This section is added to address the initial and commissioning testing programs.	Addition of this section facilitates documentation of compliance to DOE/RL-96-0006 (DOE-RL 1998b), section 4.2.8, "Pre-Operational Testing", and section 5.2.6, "Pre-Startup Safety Review", and DOE/RL-96-0003 (DOE-RL 1998a), section 4.3.2, "Contractor Input", item 13.
3.11 Operational Practices	This section is to added to address such conduct of operations considerations as shift routine and turnover, control area activities, communications, control of on-shift training, control of equipment and system status, lockout and tagout, independent verification of equipment status, logkeeping, and operational aids postings.	These items are discussed to address what is normally considered conduct of operations.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
4.7 Results of the Integrated Safety Assessment	The results for unmitigated accidents are compared to the radiological standards discussed in Integrated Safety Management Plan (ISMP) section 1.2, "Detailed Description of the Safety Approach" rather than to 10 CFR 20, "Standards for Protection Against Radiation". A full assessment of the hazardous situations that might present themselves during facility operation is provided. This includes estimates of radiological and chemical releases for this range of events. Additional details are provided on the methodology used for consequence analysis, bounding conditions, input assumptions, and accident sequences.	The standards provided in RG 3.52 were derived from 10 CFR 20, "Standards for Protection Against Radiation", which is applicable to normal operation. The nature of the accidents for the WTP requires more discussion of consequence analysis than that required of fuel fabrication facilities.
4.8 Controls for Prevention and Mitigation of Accidents	This section identifies the specific safeguards selected for protection of the facility workers, as well as safeguards selected for protection of the public and collocated workers.	The nature of the accidents for the WTP requires more discussion of consequence analysis than that required for fuel fabrication facilities.
5.0 Radiation Safety	Chapter 5.0 provides the upper-level statutory standards and program policies that ensure the radiological safety of employees, visitors, and onsite members of the public. Deviations from RG 3.52 are as follows: 1) As an US Nuclear Regulatory Commission (NRC) document, RG 3.52 references and specifies applicable portions of 10 CFR 20. Because 10 CFR 835 is the radiation safety regulation for the WTP, the focus of this section is on 10 CFR 835. 2) The implementation-level standards and guidance documents referenced in RG 3.52 is being incorporated into the Radiation Protection Plan (RPP).	Compliance with 10 CFR 835 is a requirement of the contract. The RPP required by 10 CFR 835 is required to include some of the information required of RG 3.52. There is no need to present this information in two documents.
5.1 As Low As Reasonably Achievable (ALARA) Policy and Program	RG 3.52 states that Regulatory Guide 8.10, Revision 1R (*Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable)* should be used in the development of the ALARA program. DOE guidance such as DOE G 441.1-2, *Occupational ALARA Program Guide* will also be used to develop the WTP ALARA program for normal operation.	DOE practices have proven to be successful for facilities similar to the WTP.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
5.3 Radiological Safety Standards	Section 5.3 is added to provide the radiation standards by which the program operates. The standards specifically identify regulatory exposure standards, administrative exposure control levels, and other key standards of the radiation protection program.	The contract requires compliance to the 10 CFR 800 series of nuclear safety requirements. This includes compliance to 10 CFR 835, "Occupational Radiation Protection". Section 5.3 documents the compliance to the exposure standards of those regulations that have been promulgated.
5.8 External Exposure (renumbered 5.9 from RG 3.52)	By RG 3.52, the applicant is expected to participated in the National Voluntary Laboratory Accreditation Program (NVLAP) external dosimetry. Section 5.8 allows for participation in either the NVLAP or US Department of Energy (DOE) Laboratory Accreditation Program (DOELAP) accreditation programs.	The option of participating in either the NVLAP or the DOELAP provides maximum flexibility and equivalent dosimetry program quality
5.14 Radioactive Waste Management	RG 3.52 does not require a discussion of waste management systems.	Section 5.14 is added to the SARs as the Process Hazards Analysis (PHA) completed for the WTP have identified hazards and hazardous situations with the waste management features of the facility. It is a requirement of DOE/RL-96-0003 (DOE-RL 1998a), section 4.1.2, "Contractor Input", that deliverables be tailored to the nature and level of hazards associated with its waste processing activities.
Appendix 5A Radiation Protection Program Outline	This appendix is added to address compliance to 10 CFR 835.	The contract requires compliance to the 10 CFR 800 series of nuclear safety requirements. This includes compliance to 10 CFR 835, "Occupational Radiation Protection".
Appendix 5B Environmental Radiation Protection Program Outline	This appendix is added to address compliance to the requirements of the Environmental Protection Agency (EPA) and Washington State laws and regulations.	The contract requires submittal of an outline for the environmental radiological protection plan.
Chapter 6.0 Nuclear Criticality Safety	The methodology for criticality analyses is provided in the SARs to the extent the need to perform criticality calculation is found to be appropriate. The WTP SARs provide fewer details and commitments compared to fuel fabrication facilities relative to: 1) Nuclear criticality safety organization (section 6.2.1) 2) Criticality training (section 6.2.5) 3) Specific maintenance and quality assurance provisions for criticality prevention (sections 6.2.3 and 6.2.4) 4) Audits and inspection (section 6.2.6)	RG 3.52 focuses heavily on accidental criticality which is a more significant concern for fuel fabrication facilities which have a much higher inventory and concentrations of fissile material than the WTP. See ISMP section 3.8, "Criticality Safety", for additional information.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
7.4 "Hazardous Waste Management"	Section 7.4 of the WTP SARs address all chemical inventories that are identified by the PHA as representing a significant hazard.	By section 4.2.2, "Contractor Input", of DOE/RL-96-0003 (DOE-RL 1998a), the Initial Safety Analysis Report (ISAR) is to address process safety as well as radiological and nuclear safety. The need to address all aspects of chemical safety is also a NRC requirement of RG 3.52, section 7.4, and NUREG-1513, "Integrated Safety Analysis Guidance Document", (draft) (NRC 1994). The NUREG-1513 definition of "integrated" provided in section 2.1, "Definition", makes reference to chemical safety. Specific guidance for chemical safety is provided in section 2.6.2, "Process Safety Information", of the NUREG-1513.
10.0 Environmental Protection	This chapter references the Environmental Report	Protection of the environment is addressed in a separate document.
11.0 Deactivation and Decommissioning	This chapter addresses design and operational provisions considered to facilitate deactivation and decommissioning. It does not address the financial considerations for decommissioning.	The scope of the contract (DOE-ORP 2000) is limited to design support for deactivation.

1. Standard Format and Content for the Health and Safety Sections of License Applications for Fuel Cycle Facilities, Regulatory Guide 3.52, Revision 2, draft, US Nuclear Regulatory Commission, Washington DC (NRC 1995).

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
1.1.1 Facility Description	A description of the facility design is provided in sufficient detail to demonstrate the facility design and construction requirements of the Safety Requirements Document (SRD). The details are also sufficient to support an understanding of the safety analysis provided in section 4.2, "Facility Description".	This section updates the general description of the facility design.
1.1.2 Process Description	This section describes the process design in sufficient detail to demonstrate the system and component design and fabrication requirements of the SRD are satisfied. Details on the process design sufficient to support an understanding of the safety analysis are provided in section 4.3, "Process Description".	This section updates the general description of the process design.
1.2 Institutional Information	This section provides the information required by RG 3.52, draft (NRC 1995a).	This section updates any changes in the institutional information provided in the PSAR.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
1.3 Site Description	A description of the site land use, meteorology, hydrology, geology, and seismology is provided.	This section address any existing or planned changes in land use from that provided in the PSAR. The FSAR provides any new meteorology, hydrology, geology, and seismology data made available. However, the level of detail provided for these subject areas is not significantly different between the two SARs. The FSAR summarizes data obtained during the Facility excavation that confirms the adequacy of the design. This includes the results of field and laboratory investigation of soil properties.
2.1 Organization and Administration	The Project organizational charts with a focus on the design and construction management organizations are provided. An organization chart for the operational phase is also presented. More definitive information on the roles, responsibilities, and interfaces for project management, engineering, construction management, inspections, procurement, quality assurance, records management, and nuclear safety functions is included. Section 2.1 also provides the criteria to determine minimum staffing requirements. A summary of procedures to be developed to implement the regulatory requirements addressed in this section is presented.	The section contains an update to the organizational structure of Project with a focus on operational and operational support organizations. This section also includes: 1 Title of each position that is important to public and worker safety and reporting relationship 2 Description defining qualifications, responsibilities, and authorities for each position related to safety 3 Organizational charts of the line organization and safety organization 4 Title of the individual delegated overall responsibility for the safety programs who has the authority to shut down operations if they appear to be unsafe, including independence of this authority from operational constraints 5 Lines of responsibility and authority for safety 6 Lines of communication and interfaces between organizations inside the facility 7 Availability of personnel within the safety organization to carry out the assigned function Specific information on procedure development and minimum staffing requirements is provided.
2.2 Safety Committees	Information on responsibilities, authorities, and proposed charters of safety committees, and oversight groups is provided.	This section updates information on safety committees, and oversight groups that are established following issuance of the PSAR and addresses any new safety committees that have been established.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.1 Configuration Management	This section contains specific information on: 1 Content and reference to procedures used to maintain effective configuration management of the WTP 2 Scope of identified systems, structures, and components (SSCs) and their relationship to the contents of Chapter 4.0, "Integrated Safety Analysis" 3 Description of the design information package contents to be provided to the safety analysts 4 Change control system specifics, including identification, technical and management reviews, documentation, and implementation 5 Specific physical configuration assessment, and periodic equipment performance monitoring 6 Design, installation, and testing of facility modifications 7 Revision of operating, test, calibration, surveillance, and maintenance procedures and drawings 8 Selection and control of replacement parts 9 Description of how the WTP design requirements and design basis were established and documented A summary of procedures developed to implement the regulatory requirements addressed in this section 3.1 is presented. This section also includes a draft of the unreviewed safety question process.	Specific information on the content of procedures and training developed is provided. The final unreviewed safety question process is provided.
3.2 Maintenance	A list of Safety Design Class and Safety Design Significant SSCs is provided. The maintenance implementation plan is described to such a level that maintenance philosophy and approach are evident.	The FSAR may modify the list of SSCs actions to be addressed based on safety analysis of the final design. Specific information on procedures and training developed to implement the requirements of section 3.2 is provided. In addition, the elements of the finalized maintenance implementation plan is described. Also discussed is the application of information obtained from demonstration testing and commissioning programs to the maintenance program (the latter by FSAR amendment after initial submittal).

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.3 Quality Assurance	Information related to the roles, responsibilities, and interfaces for project management, engineering, construction management, inspections, procurement, quality assurance, records management, and nuclear and process safety functions is provided. Included is the organizational structures of the quality assurance organization. The PSAR describes the quality assurance requirements of SSCs. Requirements for procedures to implement the regulatory requirements is presented.	For the FSAR, this section focuses on the quality assurance program for the operating WTP. Specific information on procedures and training developed to implement the requirements of section 3.3 is provided.
3.4 Training and Qualification	A description of the performance-based training program for operational and support personnel, including a detailed description of the training development process, is provided. The administrative process to be applied to training activities is described to a level such that the elements of the program and management's commitment to training is evident.	Details on the training and qualification program are provided. Also discussed is the application of information obtained from demonstration testing and commissioning programs (the latter by FSAR amendment after initial submittal).
3.5 Human Factors	This section documents the criteria by which human factors are considered in the facility design and operation.	This section states how human error in facility operations was taken into account in the design by facilitating correct decisions by operators and inhibiting wrong decisions. Consideration given in the design to detecting and correcting or compensating for errors is discussed.
3.6 Audits and Assessments	Information on the performance of audits and assessments is incorporated into this section.	This section is focused on audits and assessments performed during WTP operation. Specific information on procedures and training developed to implement the requirements of this section is provided.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.7 Incident Investigation	This section includes the following: 1 Provisions for establishing investigating teams 2 Functions, responsibilities, and scope of authority of investigating teams 3 Qualifications of internal and/or external investigators on investigating teams 4 A description of the procedures to ensure prompt investigation of an incident 5 Policy directives that the investigative process and the investigating team be independent of line management and that participants be assured of no retribution from participating in investigations 6 The approach proposed to determine the root cause(s) of incidents to ensure that the process is reasonable, systematic, and structured 7 Methods to ensure that corrective actions to resolve findings from incident investigations are tracked to completion 8 Identification and application of lessons learned 9 Specific reporting criteria for incident reporting during the construction phase. A summary of procedures developed to implement the regulatory requirements addressed in section 3.7 is presented.	Specific information on procedures and training developed to implement the requirements is provided. Included are specific reporting criteria for incident reporting during the operations phase.
3.8 Records Management	This section contains the organization structure and a description of the records management system, including authorities, responsibilities, and qualifications of personnel managing Environmental, Safety, and Health (ES&H) records. A summary of procedures developed to implement the regulatory requirements contained in section 3.8 is presented.	Specific information on procedures and training developed to implement the requirements is provided.
3.9 Procedures	A description of the administrative controls to ensure that work is performed in accordance with established technical standards and using approved instructions and procedures is provided.	This section describes the detailed processes of selecting activities requiring operating, emergency, and support procedures; preparing procedures; verifying and validating procedures; and reviewing and approving procedures. In addition, the program to administratively control procedures and their use is described in detail.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.10 Testing Program and Preoperational Safety Review	This section describes the analysis used to identify and define pre-operational and commissioning tests and describes tests required to ensure compliance to safety specifications. The testing program and controls are described to a level such that the testing philosophy and approach are evident. The prestart safety review approach is described to a level such that the areas to be evaluated and the evaluation approach are evident.	This section may modify the list of required safety improvement program and commissioning tests based on safety analysis of the final design. In addition, the administrative and program controls applicable to the test program are described in full.
3.11 Operational Practices	A description is provided of operational practices influenced by design details (i.e., communications systems, operational hazards associated with systems and hardware, and control area arrangements).	A description is provided of the operational practices influenced by the final design. In addition, final descriptions are provided on controls and administration of operational practices.
4.0 Integrated Safety Analysis	The methodology for hazards identification and accident analyses is described. The accident consequence analyses include margins in assumptions, boundary conditions, modeling and comparisons to acceptance criteria, as appropriate, to account for uncertainties in the design and plans for operation. Section 4.7 addresses the relationship of these uncertainties to the need to provide sufficient information in the construction authorization package to allow for issuance of the construction authorization.	Assumption used the PSAR to account for uncertainties in the design and plans for operations are removed from the FSAR analysis to the extent that these uncertainties have been resolved.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
4.2 Facility Description	In addition to providing a general description of the facility, this section discusses the basic civil/structural criteria to be applied to the design. For those structures classified as Safety Design Class, this includes the following: 1 Design codes, standards, and specifications 2 Loading criteria and load combinations 3 Design and analysis methodology 4 Structural acceptance criteria 5 Criteria for identifying testing and in-service inspection requirements 6 Material specifications 7 Special construction features This section also discusses: 1 Assumed soil properties 2 Excavation, backfill, and recompaction criteria 3 Assumed bearing capacity of the soil and the safety factor applied to this capacity 4 Expected static and dynamic building total and differential settlements. Less detail is provided for Safety Design Significant structures. Section 4.2 gives specific attention to those structures classified in section 4.8 as Safety Design Class. Structures located away from the buildings containing significant hazards and that have no relationship to nuclear or process safety are briefly described (e.g., structural design, and the contents and functions of the building) and identified on a plot plan.	The FSAR updates the facility description and basic civil/structural criteria provided in the PSAR. It follows with discussions of the results of the application of these criteria to specific features of the facility. Examples are as follows: 1 The confirmation of soil properties obtained during excavation 2 A table providing the building total and differential settlement data obtained 3 Derived soil damping values 4 The results of the soil/structure analysis 5 Developed floor response spectra and time histories 6 A list of moderate and high energy systems 7 A list of specific missile and jet impingement sources, targets, and barriers provided. Also provided are updated plan and section drawings for structures classified as Important-to-Safety. These drawings show the basic floor arrangements, location of major systems and equipment, and basic building dimensions. For those structures classified as Safety Design Class, the drawings also show key structural elements, such as panel and floor reinforcements, cell liners, leak chases, major equipment anchors, and the use of masonry walls.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
4.3 Process Description	The description of process systems includes process flow diagrams for the major systems with instrumentation, sample points, and control features noted to the extent they have been developed. Heat loads are provided for heat transfer systems important to the safety analysis. Design features and parameters important to section 4.7, "Results of the Integrated Safety Assessment", are provided. This section contains the following additional detail for each system classified as Safety Design Class: 1 The specified safety function(s) with reference to PSAR section 4.7 for the basis 2 The design basis to be applied in the development of the system design 3 Design margins to be applied 4 The criteria to be used for the development of material specifications 5 Criteria to be used to determine design limits (such as pressure and temperature) 6 Criteria to be used to identify the need for instrumentation to monitor process conditions and the design criteria for such instrumentation (e.g., application of the single-failure criterion, and testability). For many cases, the design criteria provided are those included in the SRD.	This section updates the PSAR description of process systems. Process and instrumentation diagrams are provided for major systems. In addition, for those systems classified as Safety Design Class, the FSAR describes how the design requirements provided in the PSAR are reflected in the final design. For each system classified as Safety Design Class, the following are provided: 1 The specified safety function(s) with reference to section 4.7 for the basis 2 The design basis 3 The design safety margins provided by the final design 4 Important quantitative design parameters met by the system design with their basis (e.g., heating, ventilation, and air-conditioning flow, and what established the minimum and maximum flow limits) 5 Material specifications 6 Established design limits and their basis (e.g., maximum pressure and temperature limits and what established these limits) 7 Instrumentation provided with attributes, including redundancy, diversity, in situ testability, environmental qualification, failure mode on loss of power, and the surveillance requirements as defined in section 4.8, "Controls for Prevention and Mitigation of Accidents". The means by which the monitoring requirements established in section 4.8 are also to be discussed in the FSAR. Potential adverse system interactions between systems of various design classification are addressed.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
4.7 Results of the Integrated Safety Analysis (ISA)	In addition to providing the results of the Process Hazards Analysis (PHA) and accident analysis, this section discusses the uncertainties of the PHA and accident analysis and relates these uncertainties to the required content of the construction authorization package. Section 4.7 provides the basis for the conclusion that resolution of the uncertainties will not have a significant impact on the construction authorization request. This discussion includes the following: 1 Characterization of the specific technical information that must be obtained to demonstrate acceptable resolution of the uncertainties 2 An outline and schedule of the program to resolve uncertainties 3 A discussion of the design and/or operational alternatives to resolve the uncertainties Section 4.7 of the PSAR also describes the preliminary Fire Hazard Analysis (FHA) and the consequence of each design-basis fire scenario, including the consequences in the area of origin and adjacent areas.	This section documents the resolution of any uncertainties identified in the PSAR. The FSAR describes the final FHA and all resolved uncertainties previously included in the PSAR and additional fire protection measures and equipment design.
4.8 Controls for Prevention and Mitigation of Accidents	Draft Technical Safety Requirements are included.	Final Technical Safety Requirements are included.
5.0 Radiation Safety	This chapter identifies the radiological exposure standards by which the radiation safety program is developed and the facility is operated to ensure the radiological safety of the public and workers. This chapter identifies the radiation protection criteria to be implemented in the facility design.	This chapter reflects the final facility design developed to the radiation protection criteria. It also describes the facility organization and plans for the conduct of operations. This chapter includes detail on facility operation within the radiological protection program exposure standards and other radiological protection requirements.
6.0 Criticality	The methodology for criticality analyses is provided to the extent the need to perform criticality calculation is found to be appropriate. The analyses may include margins in assumptions, bounding conditions, modeling and comparisons to the acceptance criterion, as appropriate, to account for uncertainties in the design and plans for operation.	Assumptions used in the PSAR to account for uncertainties in the design and plans for operations are removed from the FSAR criticality analysis to the extent that these uncertainties have been resolved. The FSAR describes the remaining criticality controls appropriate for the WTP.
7.0 Chemical Safety	The chapter identifies the program standards by which the chemical safety program is developed and operated to protect the public and workers against chemical hazards and hazardous situations. This chapter identifies criteria to be used for the development of chemical safety controls.	The chapter reflects the final facility design and facility organization and the developed plans for conduct of operations as related to chemical safety. This section also identifies the specific chemical safety controls to be implemented for protection of the public and workers.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
8.0 Fire Safety	This chapter describes automatic and manual fire protection features and administrative controls of the fire safety program. Also described are features of the ventilation system, building layout, and emergency egress routes important to fire safety.	Administrative controls to be implemented for the fire safety program are described, including final responsibilities of response forces, and the pre-fire plan used by firefighting personnel to suppress fires safely and effectively.
9.0 Emergency Management	This chapter identifies the applicable requirements and criteria to which the WTP Emergency Management Program are developed. A general outline of the program is presented and the relationship to the Hanford Site and local emergency management programs is discussed. Information is presented to demonstrate that the WTP staff will be able to attain an acceptable state of emergency preparedness by the time the facility becomes operational.	The FSAR discusses and references the specific emergency plan and implementing documentation prepared for the WTP. Specific aspects of all elements of the emergency preparedness program are discussed. Information is presented demonstrating the developed emergency preparedness program is compliant with applicable requirements, regulations, criteria, and guidance, and capable of responding to any operational emergency at the facility.
10.0 Environmental Protection	This chapter references the WTP Environmental Report submitted in Part A.	This chapter references the WTP Environmental Report as a new or revised Environmental Report and is not required to support the operating authorization request.
11.0 Deactivation and Decommissioning	This chapter identifies design considerations given to facilitate deactivation and decommissioning. It also discusses in general terms the planning, safety analysis, and regulatory considerations to be given to deactivation.	The chapter describes the specific design features included to facilitate deactivation and decommissioning. The level of detail for planning, safety analysis, and regulatory considerations to be given to deactivation is about the same as that provided in the PSAR. The FSAR is amended near the end of waste processing operation to provide more specific information regarding deactivation. (See Integrated Safety Management Plan Table 9-5.)

[1] *Standard Format and Content for the Health and Safety Sections of License Applications for Fuel Cycle Facilities*, Regulatory Guide 3.52, Revision 2, draft, US Nuclear Regulatory Commission, Washington, DC (NRC 1995).

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
	Introduction		Executive Summary
1.0	General Information	N/A	
1.1	Facility and Process Description		Executive Summary
1.1.1	Facility Description	2.3	Facility Overview
		2.4	Facility Structures
1.1.2	Process Description	2.5	Process Description
1.2	Institutional Information	N/A	
1.2.1	Identity and Address	1.1	Introduction
1.2.2	Activity	N/A	
1.2.3	Site Location	1.3.1	Geography

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
1.2.4	Type, Quantity, and Form of Licensed Material	3.3.2	Hazard Identification (Vol II - V)
1.3	Site Description	1.3	Site Description
1.3.1	Geography	1.3.1	Geography
1.3.2	Demography and Land Use	1.3.2	Demography and Land Use
1.3.3	Meteorology	1.4.1	Meteorology
1.3.4	Hydrology	1.4.2	Hydrology
1.3.5	Geology and Seismicity	1.4.3	Geology
2.0	Management Organization	17	Management, Organization, and Institutional Safety Provisions
2.1	Organization and Administration	17.3	Organizational Structures, Responsibilities, and Interfaces
2.1.1	Organizational Commitments, Relationships, Responsibilities, and Authorities	17.3	Organizational Structures, Responsibilities, and Interfaces
2.1.2	Management Controls	17.4	Safety Management Policies and Programs
2.2	Safety Committees	17.4.2	Safety Review and Performance Assessments
3.0	Conduct of Operations	11	Operational Safety
3.1	Configuration Management	17.4.3	Configuration Management
3.2	Maintenance	10.5	Maintenance
3.3	Quality Assurance	14	Quality Assurance
3.3.1	Management Commitment for QA Program	14	Quality Assurance
3.3.2	Scope of QA Program	14	Quality Assurance
3.3.3	Organizational Responsibility	14	Quality Assurance
3.3.4	QA Program Description	14	Quality Assurance
3.3.5	Graded QA Approach	14	Quality Assurance
3.3.6	Application of Graded QA to SSCs and Activities	14	Quality Assurance
3.4	Training and Qualification	12	Procedures and Training
3.4.1	Organization and Management of the Training System	12.4	Training Program
3.4.2	Trainee Selection	12.4	Training Program
3.4.3	Conduct of Needs/Job Analysis and Identification of Tasks	12.4	Training Program
3.4.4	Development of Learning Objectives as the Basis for Training	12.4	Training Program

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
3.4.5	Organization of Instruction Using Lesson Plans and Other Training Guides	12.4	Training Program
3.4.6	Evaluation of Trainee Mastery of Learning Objectives	12.4	Training Program
3.4.7	Conduct of On-The-Job Training	12.4	Training Program
3.4.8	Systematic Evaluation of Training Effectiveness	12.4	Training Program
3.5	Human Factors	13	Human Factors
3.5.1	Organization and Administration	13.3	Scope of Human Factors Process
3.5.2	Human Factors and Assessment of the Correction of Deficiencies	13.4	Human Factors Program
3.6	Audits and Assessments	17.4.2	Safety Review and Performance Assessment
3.7	Incident Investigations	13.4	Human Factors Program
3.8	Records Management	17.4.4	Document Control and Records Management
3.8.1	Organization and Administration	17.4.4	Document Control and Records Management
3.8.2	Types of Records	17.4.4	Document Control and Records Management
3.8.3	Record Handling Procedures	17.4.4	Document Control and Records Management
3.8.4	Record Storage and Protection	17.4.4	Document Control and Records Management
3.9	Procedures	12.3	Procedures Program
3.10	Testing Program and Preoperational Safety Review	10.3	Commissioning
3.11	Operational Practices	11.3	Conduct of Operations
4.0	Integrated Safety Analysis	3	Hazard and Accident Analysis
4.1	Site Description	1.3	Site Description
4.2	Facility Description	2.3 2.4	Facility Overview Facility Structures
4.3	Process Description	2.5	Process Description
4.4	Process Safety Information	3.3.3.X	Hazard Evaluation
4.5	Training and Qualifications of ISA Team	3.3.1	Identification of Work
4.6	ISA Methods	3.X	Hazard and Accident Analysis
4.7	Results of the Integrated Safety Assessment	3.3.3 (Vol II - V)	Development of Control Strategies

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
4.8	Controls for Prevention and Mitigation of Accidents	3.4 (Vol II - V)	Accident Analysis Methodology
4.9	Administrative Control of the ISA		Executive Summary
5.0	Radiation Safety	7	Radiation Protection
5.1	As Low As Reasonably Achievable (ALARA) Policy	7.1	Introduction
5.2	Organizational Relationships and Personnel Qualifications	7.1	Introduction
5.3	Radiological Safety Procedures and Radiological Work Permits (RWPs)	7.1	Introduction
5.4	Training	7.1	Introduction
5.5	Ventilation systems	2.6	Confinement Systems
5.6	Air Sampling	7.1	Introduction
5.7	Contamination Control	7.1	Introduction
5.8	External Exposure	7.1	Introduction
5.9	Internal Exposure	7.1	Introduction
5.10	Summing Internal and External Exposure	7.1	Introduction
5.11	Respiratory Protection	7.1	Introduction
5.12	Instrumentation	7.1	Introduction
5.13	Integrated Safety Analysis	3.4	Accident Analysis Methodology
5.14	Radioactive Waste Management	8	Hazardous Material Protection
6.0	Nuclear Criticality Safety	6	Criticality Safety Program
6.1	NCS Technical Practices	6.3 6.4	Criticality Limits and Concerns Criticality Controls
6.1.1	Process Analysis from the Integrated Safety Analysis	6.4.6	Application of Double Contingency Principle
6.1.2	NCS Evaluations	6.4.6	Application of Double Contingency Principle
6.1.3	NCS Limits	6.4.3	Administrative Controls
6.1.4	Validation and Use of Analytical Methods	6.4.4	Methodology for Determining Criticality Limits
6.1.5	NCS Control Methods	6.4.3	Administrative Controls
6.1.6	Criticality Accident Alarm System	6.6	Criticality Instrumentation
6.2	Administrative Practices	6.5	Criticality Protection Program
6.2.1	NCS Organizational Responsibilities	6.5	Criticality Protection Program
6.2.2	Configuration Management	17.4.3.2	Configuration Management Process
6.2.3	Maintenance	6.4.3	Administrative Controls

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
6.2.4	Quality Assurance (QA)	6.4.3	Administrative Controls
6.2.5	Training	6.5.4	Criticality Safety Training and Qualifications
6.2.6	Operational Inspections, Audits, Assessments, and Investigations	6.5.5	Criticality Safety Training and Qualifications
6.2.7	Written Operating Procedures	6.5.3	Administrative Controls
6.2.8	Materials Control for NCS	N/A	
6.2.9	Emergency Preparedness	6.6	Criticality Instrumentation
7.0	Chemical Safety	8	Hazardous Material Protection
7.1	Chemical Safety Responsibility	8.3	Hazardous Material Protection and Organization
7.2	Chemical Safety Approach	8.6	Hazardous Material Exposure Control
7.3	Chemical Safety Controls	8.6	Hazardous Material Exposure Control
7.4	Hazardous Waste Management	8.6	Hazardous Material Exposure Control
8.0	Fire Safety	N/A	
8.1	Organization and Conduct of Operations	18.3	Organization and Management
8.1.1	Organization and Management	18.3	Organization and Management
8.1.2	Training and Qualifications	18.4	Training and Qualifications
8.1.3	Fire Prevention Program	18.5	Fire Prevention Program
8.2	Fire Protection Features and Systems	18.6	Fire Protection Features and Systems
8.3	Manual Fire-Fighting Capability	18.7	Manual Fire-Fighting Capability
8.4	Fire Hazard Analysis	18.8	Fire Hazard Analysis
8.5	References	N/A	
9.0	Emergency Management	15	Emergency Preparedness
9.1	Description of On-Site and Off-Site Emergency Facilities	15.4.4	Emergency Facilities and Equipment
9.2	Types of Accidents	3.3.3	Development of Control Strategies
9.3	Classification of Accidents	3.3.3	Development of Control Strategies
9.4	Detection of Accidents	15.4	Emergency Preparedness Planning
9.5	Mitigation of Consequences	15.4	Emergency Preparedness Planning
9.6	Assessment of Releases	15.4	Emergency Preparedness Planning
9.7	Responsibilities of Licensee and Other Organizational Personnel	15.4	Emergency Preparedness Planning
9.8	Notification and Coordination	15.4	Emergency Preparedness Planning
9.9	Description of the Emergency Operational Center	15.4	Emergency Preparedness Planning

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
9.10	Information to be Communicated and the Parties to be Contacted	15.4	Emergency Preparedness Planning
9.11	Public Notification	15.4	Emergency Preparedness Planning
9.12	Training	15.4	Emergency Preparedness Planning
9.13	Procedures for Safe Shutdown and Recovery	15.3 15.4	Scope of Emergency Preparedness Program Emergency Preparedness Planning
9.14	Drills and Exercises	15.4	Emergency Preparedness Planning
9.15	Procedures for Identifying, Locating, and Controlling Hazardous Chemicals	15.3 15.4	Scope of Emergency Preparedness Program Emergency Preparedness Planning
9.16	Responsibilities for Developing and Maintaining Current the Emergency Program and Its Procedures	15.3 15.4	Scope of Emergency Preparedness Program Emergency Preparedness Planning
10.0	Environmental Protection	9.1	Introduction
10.1	Environmental Report	N/A	
10.1.1	Description of Proposed Action	N/A	
10.1.2	Purpose of Proposed Action	N/A	
10.1.3	Description of Affected Environment	N/A	
10.1.4	Discussion of Considerations	N/A	
10.1.5	Analysis of Environmental Effects of Proposed Action and Alternatives	N/A	
10.1.6	Federal and State Environmental Requirements	9.1.1 9.4	Permit Overview Radioactive and Hazardous Waste Processes
10.2	Environmental Safety Program	9.3	Radioactive and Hazardous Waste Management Program and Organization
10.2.1	Features for Contamination Control	2.3	Facility Overview
10.2.2	Environmental Monitoring Program	9.1 9.4	Introduction Radioactive and Hazardous Waste Processes
10.2.3	Emergency Plan	15.X	
10.2.4	Maintenance and Surveillance	9.3	Radioactive and Hazardous Waste Management Program and Organization
10.2.5	Configuration Management	17.4.3.2	Configuration Management Process
10.2.6	Organization and Management	9.3.1 17	Program Summary Management, Organization, and Institutional Safety Provisions
10.2.7	Quality Assurance	14.1	Introduction

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
10.2.8	Training	9.3 9.7 (ERPP)	Radioactive and Hazardous Waste Management Program and Organization Environmental Radiological Protection Plan (ERPP)
10.2.9	Event Notification and Reporting	9.1.1	Permit Overview
10.2.10	Bibliography	9.8	References
11.0	Decommissioining	16	Deactivation and Decommissioning
11.1	Conceptual Decommissioning Plan	16.1	Introduction
11.1.1	Information for Conceptual Decommissioning Plan	16.4	Deactivation Requirements
11.1.2	Information for Total or Partial Cessation of Operations	16.5	Transition Readiness
11.1.3	Bibliography	16.7	References
11.1.4	Appendix A: Cost Estimating Tables	N/A	
11.2	Decommissioning Funding Plan and Financial Assurance Mechanisms	N/A	
11.2.1	Decommissioning Cost Estimate	N/A	
11.2.2	Financial Assurance Mechanism(s)	N/A	
11.2.3	Updating the Cost Estimate and Funding Level	N/A	
11.2.4	Bibliography	N/A	
11.2.5	Appendix A: Sample Sight Draft	N/A	
Appendix A	Radiation Protection Program Outline	7 RPP	Radiation Protection
Appendix B	Environmental Radiation Protection Program Outline	9.5 ERPP	Environmental Radiation Protection Plan (ERPP)

The SARs should include multiple volumes. Volume I should provide information that is applicable to more than one of the facilities (e.g., Pretreatment, Low-Activity Waste Vitrification, High-Level Waste Vitrification, and Balance of Facilities). Other volumes should be facility specific and contain, at a minimum, chapters 2, 3, 4, and 5.

Executive Summary

- **E.1 Facility Background and Mission**
- **E.2 Facility Overview**
- **E.3 Facility Hazard Classification**
- **E.4 Safety Analysis Overview**
- **E.5 Organization**
- **E.6 Safety Analysis Conclusions**
- **E.7 SAR Organization**
- **E.8 Summary of Significant Changes from the Preliminary Safety Analysis Report (FSAR stage)**

1 Site Characteristics

- **1.1 Introduction**
- **1.2 Requirements**
- **1.3 Site Description**
- **1.4 Environmental Description**
- **1.5 Natural Phenomena Hazards**
- **1.6 External Man-Made Threats**
- **1.7 Nearby Facilities**
- **1.8 References**

2 Facility Description

- **2.1 Introduction**
- **2.2 Requirements**
- **2.3 Facility Overview**
- **2.4 Facility Structures**
- **2.5 Process Description**
- **2.6 Confinement Systems**
- **2.7 Safety Support Systems**
- **2.8 Utility Distribution Systems**
- **2.9 Auxilliary Systems and Support Facilities**
- **2.10 References**

17 Management, Organization, and Institutional Safety Provisions

- **17.1 Introduction**
- **17.2 Requirements**
- **17.3 Organizational Structures, Responsibilities, and Interfaces**
- **17.4 Safety Management Policies and Programs**
- **17.5 References**

18 Fire Protection

- **18.1 Introduction**
- **18.2 Requirements**
- **18.3 Organization and Management**
- **18.4 Training and Qualification**
- **18.5 Fire Prevention Program**
- **18.6 Fire Protection Features and Systems**
- **18.7 Manual Fire-Fighting Capability**
- **18.8 Fire Hazard Analysis**
- **18.9 Fire Protection for Filter Plenums**

4.0 References

BNI, 2001. *Safety Requirements Document Volume II*, 24590-WTP-SRD-ESH-01-001-02. Bechtel National, Inc., Richland, Washington.

DOE-ORP, 2000. Contract No. DE-AC27-01RV14136, December 2000. US Department of Energy, Office of River Protection, Richland, Washington.

DOE-RL, 1998a. *DOE Regulatory Process for Radiological, Nuclear, and Process Safety for TWRS Privatization Contractors*, DOE/RL-96-0003, Revision 1. US Department of Energy, Richland Operations Office, Richland, Washington.

DOE-RL, 1998b. *Top-Level Radiological, Nucelar, and Process Safety Standards and Principles for TWRS Privatization Contractors*, DOE/RL-96-0006, Revision 1. US Department of Energy, Richland Operations Office, Richland, Washington.

NRC, 1994. *Integrated Safety Analysis Guidance Document*, draft NUREG-1513, distributed 8 August 1994, US Nuclear Regulatory Commission, Washington, DC.

NRC, 1995. Regulatory Guide 3.52, *Standard Format and Content for the Heatlh and Safety Sections of License Applications for Fuel Cycle Facilities*, draft Revision 2, US Nuclear Regulatory Commission, Washington, DC.

Appendix H

Ad Hoc Implementing Standard for Erosion/Corrosion and Assessments

CONTENTS

1.0 Introduction

The Waste Treatment and Immobilization Plant (WTP) will treat a wide range of radioactive wastes. As the waste enters the plant, all of the waste will be alkaline. Part of the treatment process may, however, require the waste stream to be acidified. Whether the waste is acidic or alkaline, the materials of construction will be subject to corrosion.

Corrosion is a degradation process affected by many parameters such as temperature, chemistry, flow rate, stress, and/or vibration. The degradation may be electrochemical in nature, chemical, mechanical, or a combination of all. Often combinations of parameters act synergistically, sometimes reducing the corrosion rate but often accelerating the rate or changing the mechanism.

Erosion is the removal from the surface by the action of particles in a moving liquid or gas or liquid particles in a moving gas. In WTP, many of the waste treatment streams contain solids. Others, such as off-gas lines, may contain liquid or solid particles. Erosion is a function of the fluid velocity and particle size, shape, and relative hardness. Erosion-corrosion is corrosion exacerbated by the erosive removal of protective layers, which allows corrosion to proceed at a high rate.

Evaluation, selection, and establishment of corrosion and erosion control measures begin with design and are implemented during construction and maintained during operation. Assessments are performed to ensure that vessel and piping systems have sufficient structural integrity and are acceptable for the storing and treatment of radioactive and/or chemical materials.

2.0 Corrosion Evaluations and Material Selection

Material selection begins with the chemistry conditions enveloping the expected process stream conditions. Additional information may include off-normal or accident conditions. Process information such as chemistry conditions, temperature ranges, fluid velocities, and radiation fields are determined for each vessel and associated piping. External conditions are also determined. Various materials are evaluated for general corrosion, pitting corrosion, stress corrosion cracking, crevice corrosion, end grain corrosion, corrosion at welds, microbiologically induced corrosion, corrosion fatigue, vapor phase corrosion, erosion, galling, fretting, wear, galvanic corrosion, cavitation damage, and creep. The acceptable materials are identified, and the least cost acceptable material is generally selected. General corrosion rates are derived from the literature, laboratory investigations, and experience at other plants, and a general corrosion allowance for a 40 year life is specified.

The process chemistry conditions for an evaluated important to safety (ITS) component are provided and documented on a Material Selection Data Sheet. The information is used in the preparation of the Corrosion Evaluation, which includes the process chemical conditions, corrosion analyses, material selected, corrosion allowance, and operating limitations. The Corrosion Evaluation is prepared by a metallurgist and checked by a corrosion specialist. Operation limitations due to the material selected are identified by Engineering and are checked by Operations. The Corrosion Evaluation is further reviewed by a materials and engineering technology specialist and by a corporate materials specialist to provide adequate assurance that the correct material has been chosen.

3.0 Corrosion and Erosion Mechanisms and Solutions

3.1 General Corrosion

General (uniform) corrosion rates for materials of construction on this project have been derived from the literature, laboratory investigations, and plant experience. These data are used to set corrosion allowances for vessels or piping manufactured from the specified grades of material. The general/uniform corrosion rates of the selected austenitic stainless steels and the various Ni/Cr/Mo alloys are less than 1.0 mpy.

Parameters that affect general corrosion include the conductivity of the solution, temperature, velocity (whether reactants are brought to the surface or corrosion products are removed), pH, redox potential, and concentrations of reactants and products. The effect is of each of these parameters on the corrosion rate depends on the particular system and the operable corrosion mechanism.

Once a material is selected based on considered corrosion mechanisms, a general corrosion rate shall be specified based on the process chemistry and parameters and any operating restrictions are specified to keep the chemistry and parameters within the expected ranges.

3.2 Pitting Corrosion

Pitting corrosion can take place in austenitic stainless steels, or other alloys that passivate, and usually occurs in the presence of chlorides or sulfates. However, the efficacy of these ions in promoting pitting depends on the presence of other ions, particularly nitrates and, in high radiation fields radiation generated species.

Where pitting could be a potential source of corrosion, a more resistant material than 304L such as, 316L, 6 % Mo, or Ni/Cr/Mo alloys shall be used.

3.3 End Grain Corrosion

End grain corrosion is preferential corrosion, which occurs along the worked direction of wrought stainless steels exposed to highly oxidizing acid conditions as well as in other alloys under the "suitable" conditions. This is generally not a problem unless end grains are exposed in a highly oxidizing acid condition at high temperatures.

The exposure of end grains to highly oxidizing acid conditions at high temperatures shall be avoided

3.4 Stress Corrosion Cracking

Most metals and particularly alloys, including stainless steels and the nickel base alloys, can suffer stress corrosion cracking (SCC). SCC is a phenomenon which occurs when the appropriate stress is applied to the metal, a conducive environment is present, and the metal is susceptible.

For a given alloy there are generally only a few agents that will trigger cracking, three of the more common being chloride, hydroxide, and nitrate.

The use of low carbon "L grade" alloys should prevent intergranular stress corrosion cracking (IGSCC). Caustic SCC occurs in carbon steel and 300 series stainless steels at temperatures greater than about 140 °F. Nickel rich alloys tend to be more resistant.

Where SCC could be a potential source of corrosion, low carbon alloys such as 304L or 316L or more resistant alloys such as 6 % Mo and Ni/Cr/Mo alloys shall be used. It is also important to ensure that stainless steel is not contaminated with carbon if welded directly to the stainless steel structure. Otherwise, it is possible for the stainless steel to be more susceptible to stress corrosion cracking and other forms of intergranular attack.

3.5 Crevice Corrosion

Crevice corrosion is a form of localized corrosion that can occur within crevices or at shielded surfaces where a stagnant solution is present, e.g., at metal/metal or metal/non-metal junctions such as under bolts, gaskets and valve seats. The presence of solid precipitates/sludges can also create crevice corrosion conditions. Crevice corrosion is similar to pitting in mechanism, though generally not so rapidly debilitating. It can, however, lead to pitting or stress corrosion cracking.

In general, crevices are avoided in highly oxidizing situations. Where crevice corrosion could be a potential source of corrosion, low carbon alloys such as 304L or 316L or more resistant alloys such as 6 % Mo and Ni/Cr/Mo alloys shall be used.

3.6 Corrosion at Welds

Laboratory investigations and plant experience indicate that, providing correct weld procedures are followed, no preferential corrosion of weld beads or heat affected zones occurs in nitric acid based streams. Thus, no additional corrosion allowance is made for weld bead corrosion. The alloys most commonly used on the project, alloys, 304L, 316L, 6 % Mo, and C-22, do not suffer from this form of knife line corrosion and this failure mechanism is not relevant for systems built from them.

3.7 Microbiologically Induced Corrosion (MIC)

Typically, MIC is not observed in operating systems, with the exception of cooling water systems. To minimize the potential of MIC only treated process water, potable water, or deionized (demineralized) water shall be used. Flushing and hydrotest water shall be drained and not be left standing in the pipe after the completion of testing.

3.8 Fatigue/Corrosion Fatigue

Fatigue is the phenomenon leading to fracture under cyclic stresses that have a maximum value of less than the tensile strength of the material. Corrosion fatigue is fatigue exacerbated by corrosion concurrent with or subsequent to the application of the stress.

The vessels and piping shall be designed to accommodate the expected fatigue cycles over the 40 year design life.

3.9 Vapor Phase Corrosion

Conditions in the vapor phase and at the vapor/liquid interface can be significantly different than in the liquid phase.

The corrosion in these vapor regions may be different from that in the bulk liquor and shall be considered in specifying corrosion allowances.

3.10 Erosion

This is the removal of material from a metal surface by the action of particles in a moving liquid or liquid particles in a moving gas/vapor. In the WTP, many of the streams contain solids, for example, the waste and glass-former supply lines. Others, such as steam lines, may contain other liquid or solid particles. Erosion is a function of the fluid velocity, particle size, shape, and the relative hardness of the particles to that of the material of construction.

Velocities above about 10 fps for slurries shall be specifically evaluated. The typical velocity in the lines is less than about 8 fps. Combined with the softness of the Hanford waste, little erosion is expected.

In areas where glass formers are present, a hard overlay (Stellite) shall be used to protect vessels and piping shall have a larger erosion allowance.

3.11 Galling of Moving Surfaces

Where two metals are moving in contact with each other without lubrication, there is a risk of damage to their surfaces.

Where galling could occur, a material grade, such as UNS S21800, which is less susceptible to galling, shall be used for at least one of the components or the use of dry lubricants or metallic coatings shall be used.

3.12 Fretting/Wear

Fretting results from the rubbing of two contacting surfaces. Fretting occurs at low amplitudes and can result in pit-type defects.

Where fretting or wear is a potential issue, such as in pipes passing through baffle plates, an appropriate additional corrosion allowance shall be added.

3.13 Galvanic Corrosion

When a metal is immersed in a liquid it will establish a corrosion potential or rest potential at which the rate of anodic reaction is equal to the rate of cathodic reaction. When two dissimilar metals are placed in electrical contact in such a solution, an electrochemical cell will be set up and the difference in their rest potentials will cause a current to flow between them.

In the WTP, though several alloys may be used in a given vessel, they often will be similar and corrosion potential differences may not be great. This similarity may cease, for example, in a crevice where one component may become active and corrode severely. Due to the use of steam ejectors and heated tanks, there are opportunities for the presence of thermogalvanic corrosion cells to be set up. If two portions of the same component are at different temperatures, the warmer section often becomes the anode and corrodes.

Galvanic corrosion protection shall be provided where required.

3.14 Cavitation Damage

This is caused by the formation and collapse of vapor bubbles in a liquid near a metal surface. It is not usually found in reprocessing plants but the possibility for cavitation damage exists in high velocity fluids, such as those found in fluidic devices or centrifugal pumps. Cavitation has not been encountered in fluidic devices nor in pumps when the devices are designed and operated under proper conditions.

Pumping systems and agitators shall be designed to minimize cavitation. The velocity in copper alloys, in hot water shall be less than 1.5 fps and in cold water shall be less than 4 to 6 fps to minimize cavitation.

3.15 Creep

Creep is the continuous increasing deformation of a material over time under a constant load. It is only experienced in chemical plants operating at high temperatures. The potential sites for creep in the WTP are in the thermal oxidizer and at the melters.

The high temperature vessels and piping shall be designed to allow for creep over the life of the component.

4.0 Corrosion/Erosion Allowance

Vessels and piping can be classified into the following groups:

- Vessels and piping in which the corrosion rates can be definitely established using information available regarding the chemical characteristics of the substances contained. Where the corrosion rate is closely predictable, a corrosion allowance at least equal to the expected corrosion loss over a 40 year design life shall be specified.
- Vessels and piping in which the corrosion rates are known to be relatively high and are either variable or indeterminate in magnitude. Where the corrosion rates are known, a reasonable corrosion allowance, which includes any uncertainty in the corrosion rate, shall be specified.
- Vessels and piping in which the corrosion rates are indeterminate and are known to be relatively low. Where the corrosion rates are indeterminate but expected to be low, a minimum standard corrosion allowance (typically 0.04 inch over 40 years) shall be specified.

- Vessels and piping in which corrosion effects are known to be negligible or entirely absent. When corrosion effects can be shown to be negligible or entirely absent, no corrosion allowance need be specified.

Where the solids content is greater than 4 % by weight, a minimum corrosion/erosion allowance of 0.125 inch shall be provided or hardfacing shall be provided in areas of high velocity.

5.0 Vessel and Piping Assessments

There are four general types of processes, assessments, and inspections for ITS vessel and piping systems: design process, installation inspections, routine inspections, and integrity assessments.

The design process will ensure that an ITS vessel and associated piping systems have sufficient structural integrity and are acceptable for the storing and treatment of radioactive and/or chemical materials. The design process will ensure that the vessel and associated piping have sufficient structural integrity and are acceptable for performing their safety functions. Part of this design process includes the review of factors affecting the potential for corrosion, corrosion protection systems, materials selection report, and associated corrosion evaluations. The design process will ensure that the foundations, structural supports, seams, connections, and pressure controls are adequately designed. It will also ensure that the vessel or piping system has sufficient structural strength, compatibility with the process flow stream, and corrosion protection to ensure that it will not collapse, rupture, or fail. As part of the design process, a corrosion expert will make recommendations for corrosion protection of any external metallic components in contact with soil.

The following types of information will be reviewed to determine whether the ITS vessel and associated piping system are adequately designed:

- Design drawings
- Specifications
- Mechanical data sheets
- Piping class sheets
- Layout drawings
- Isometric drawings
- Stress analyses
- Structural calculations
- Cathodic protection design documentation
- Secondary containment drawings
- Process stream characteristics
- Pressure control systems
- Piping and instrument diagrams

- Materials selection report
- Associated corrosion evaluations

Installation inspections of ITS vessels and piping will include:

- Placement of shop and field erected vessels
- Installation of secondary containment liners
- Installation of piping, piping supports, ancillary equipment, and in-line components
- Installation of cathodic protection when required
- Non-destructive examination where required
- Visual and pressure testing
- Tightness testing prior to placing the system in service

Routine inspections for ITS vessels and piping systems (such as visual inspections, camera inspections, or sump monitoring) are performed where practicable to ensure waste has not leaked out of the piping system. Jumpers can be inspected to determine if they have been subjected to corrosion or erosion damage.

Periodic integrity assessments will be performed where practicable. The periodic integrity assessments of vessels and piping will include as a minimum the review of applicable process chemistry and operating conditions over the period to ensure that they have stayed within the specified ranges and the determination of the effect on corrosion or erosion of any deviations from the specified ranges.

Where consistent with keeping the radiation exposure as low as reasonably achievable (ALARA), the periodic integrity assessments of piping and vessels may include non-destructive examination (NDE) of welds or determination of wall thickness in order to detect potential degradation in selected accessible systems. Identifying and evaluating potential degradation mechanisms will help identify areas where additional inspection may be required. However, adjustments to inspection strategy should account for consequences of a failure as well. Other factors include material of construction, design conditions relative of operating conditions, design codes and standards used, effectiveness of corrosion monitoring programs, and quality of maintenance programs.

An in-service inspection description as to where baseline measurements of welds or wall thicknesses should be taken shall be made available to DOE 6 months prior to hot commissioning to provide information that can be used to create an in-service inspection plan. Among the considerations to be considered when locating in-service inspection points are:

- Material of construction
- Corrosive characteristics of the contained substance
- Erosive characteristics of the contained substance
- The velocity or turbulence of contained substance at the point of inspection
- Scope of information to be obtained (representative of other vessels or piping)
- Access to the point (either manually or remotely)
- Jumpers may be used where representative of the vessel or piping characteristics

- Inspection methods, scope, tools and techniques that can be used
- Radiation exposure to personnel performing the inspection

In-service inspection shall be made at the selected points during routine or maintenance outages where practicable and within 10 years after start of hot operations. Process vessels and piping inspections shall be categorized into different classes. Vessels and piping with higher potential for corrosion or erosion shall be inspected within 7 years after hot commissioning. Other vessels and piping with a lower potential for corrosion and erosion shall be inspected within 10 years after hot commissioning. Subsequent inspection intervals may be reduced or increased based on process conditions, operating history, inspection results, and the expected remaining corrosion life of the vessel or piping.

6.0 Inaccessible Areas

The WTP design incorporates the "black cell" concept as a key part of the facility design for the Pretreatment (PT) and High Level Waste (HLW) facilities. This entails locating certain equipment in shielded cells for which no maintenance or entry is planned for the 40-year design life of the plant. Key to the approach is the limitation of equipment in the cell to types that require no maintenance. Thus the contents of the black cells are limited to vessels and associated pumping, mixing, and sampling systems. These systems employ fluidics (air-driven pulse jet mixers (PJMs) and reverse flow diverter (RFD) pumps with no moving parts) rather than motor-driven pumps or mechanical agitators to avoid equipment requiring maintenance or repair. In addition most of the fluid systems in WTP are low pressure, low temperature systems which have low working stresses. In these areas of WTP, detailed in-service inspection of vessels and piping during operation is impractical since access has not been provided or the radiation levels are too high to permit personnel access. In order to ensure that the piping and vessels in these areas are adequately designed and fabricated to last for a design life of 40 years without in-service inspection the following features have been included. Some of these features are included in the "black cell" vessels and piping systems to achieve a comparable reliability with systems that are expected to last over 40 year design life without failure.

- Correct Material Selection – Materials are selected and evaluated to ensure that they are compatible with the expected operating conditions (including temperature, pH, and chemistry) and will last for a design life of 40 years.
- Adequate Corrosion Allowance – The minimum general corrosion allowance for a design life of 40 years is determined based on the expected corrosion rate at the operating conditions.
- High Quality Assurance Requirements – For vessels containing significant quantities of radioactive waste, quality assurance program requirements for nuclear facilities (NQA-1) are specified.
- Vessel Design – The vessels are designed, fabricated, installed, and tested to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section VIII, Division 1, which is the industry standard for reliable vessels.
- Piping Design – The piping is designed, fabricated, installed, and tested to ASME B31.3, which is the petroleum refinery and chemical industry standard for reliable piping.
- Redundant Components – Where appropriate, redundant system components are installed in the cells and in the vessels; specifically, spare RFD pumps and (PJMs) are included in the design. Once a

failure is identified, the component can be replaced by a redundant unit. In some cases bulges have been provided for additional access to redundant components.

- Flushing Provisions – Connections are provided where required to permit flushing of potentially corrosive deposits and unplugging of fluidic pumps.
- Fatigue Analyses – Fatigue analyses are performed on the vessels and piping in accordance with the design codes to ensure that they will last for the number of expected cycles during operation over a design life of 40 years.
- Traceability of Materials – Traceability (such as identification of the item to applicable specification and grade of material, heat, batch, lot, part, or serial number or specified inspection, test, or other records) is required when specified by codes, standards, or specifications.
- Control of Welding Processes – Acceptable welding processes are defined in welding specifications used for vessels and piping.
- Positive Material Identification – Positive Material Identification (PMI) is used to check to ensure that the correct material has been used in shop fabricated vessels and piping and in selected field pipe welds where corrosion is a concern.
- Volumetric Inspection – Full volumetric inspection of the welds in the primary confinement boundary of vessels and of the girth welds in process piping is performed to ensure that weld defects are discovered and repaired.
- Hydrostatic and Pneumatic Tests – Hydrostatic or pneumatic tests will be used to ensure that the systems are leak tight prior to startup.
- Cold Chemical Testing – Cold chemical testing with simulants will be performed during startup testing which will ensure that the materials selected are compatible with the expected operating conditions.
- Monitoring of Process Operating Conditions – During operations, samples of the process flow streams will be taken periodically to ensure process conditions are within the design conditions. Also, indications may be available to measure process parameters.

During operation, the sump levels in these areas will be monitored, and if leakage is detected an assessment will be made to determine if the source of the leakage can be identified.

Appendix I

Ad Hoc Implementing Standard for Project Integrated Safety Management Approach

CONTENTS

Section **Page**

1.0 Project Integrated Safety Management Approach

The WTP Project Contractor's safety approach is implemented with the recognition that the defined work for processing and immobilizing Hanford tank waste involves inherent radiological and chemical hazards from which hazardous situations may arise. Throughout this implementing standard, safety refers to radiological, nuclear and process safety with the scope of the WTP Project Authorization Basis. The WTP Project Contractor is committed to integrating the development of safety criteria and design requirements, the hazard analysis and accident analysis process, and the facility design to minimize the risk associated with these hazards and hazardous situations. The WTP Project Contractor accepts responsibility for the safety of the WTP and for adequate protection of the health and safety of the public, worker safety, environmental protection, and compliance with applicable laws and regulations.

The safety approach for the WTP Project is based on applying best industry practices and cost-effective processes that come from successful and safe operation in the commercial and DOE nuclear environment and the chemical process industry. The purpose of the safety approach is to achieve the following objectives.

1) Ensure an adequate level of safety at the facility for the workers and the public.
2) Comply with applicable laws and regulations.
3) Conform to top-level safety standards and principles stipulated by the U.S. Department of Energy (DOE/RL-96-0006, Revision 2, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for the RPP Waste Treatment Plant Contractor*).

See SRD Volume II, Appendix A for the detailed description of the ISM process defined by DOE/RL-96-0004, Revision 2, *Process for Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for the RPP Waste Treatment Plant Contractor.*

Procedures are one tool by which compliance with requirements is ensured during the design, construction, commissioning, operation, and deactivation of the project. All activities that may affect safety of the public and workers are performed in accordance with step-by-step instruction provided in procedures. The range of activities covered in procedures includes, but is not limited to:

1) Design control
2) Procurement activities
3) Construction activities
4) Monitoring contractors
5) Identification and resolution of nonconforming conditions
6) Operations and maintenance
7) Emergency plan implementing procedures

2.0 Safety Responsibilities

Safety responsibilities are assigned to and by the WTP Project Manager. The roles assigned to organizations are provided in section 6.0, "Organization Roles, Responsibilities, and Authorities". The overall, general roles, responsibilities, and authorities assigned to WTP Project organization managers are provided in 24590-WTP-QAM-QA-01-001, *Quality Assurance Manual* for the Design, Construction, and Commissioning (DC&C) phase of the Project.

In addition, by these assignments, assurance is provided that the roles identified in the WTP Project Safety Analysis Report (SAR) are carried out.

The WTP facility design is based on the design and operational experience gained at other nuclear and chemical facilities. As such, the potential hazards are well understood and lessons learned from earlier facilities are applied.

3.0 Authorization Basis

In this section, the content, control, and update of the authorization basis are discussed. The authorization basis is the composite of information provided by the WTP Project Contractor in response to radiological, nuclear, and process safety requirements that is the basis on which the DOE grants permission to perform regulated activities related to WTP radiological, nuclear, and process safety. The authorization basis applies to the WTP project. Compliance to a standard which is included in Volume II of the SRD means that all mandatory statements (shall/will/must) applicable to nuclear, radiological, or process safety are implemented or deviations justified and approved by the DOE. Compliance with non-mandatory statements (should/may) are not required; but are reviewed and considered for each standard on an individual basis. This review is documented. Compliance to statements not applicable to nuclear, radiological, or process safety may in many cases be required to ensure compliance to regulations outside the scope of the DOE review (e.g., environmental protection); however, if no other regulatory entity requires compliance via the standard, compliance is not required to be reviewed on an individual basis.

3.1 Content of the Authorization Basis

The authorization basis for WTP includes the DOE-approved documentation. This documentation includes that information submitted in connection with a request for Standards Approval, a request for Construction Authorization, or a request for Operations Authorization as described in DOE/RL-96-0003, Revision 2, *DOE Process for Radiological, Nuclear, and Process Safety Regulation of the RPP Waste Treatment Plant Contractor*, and any other information submitted by the WTP Project Contractor in connection with these requests. Amendments to this information may be in the form of revisions to the previously submitted documents, or new information that supplements previously submitted information. The authorization basis begins at the Standards Approval regulatory action and continues throughout the design, construction, commissioning, operation, and deactivation of the WTP.

Other documents generated by the regulator or the WTP Project Contractor may become part of the authorization basis for the Project. This includes correspondence concerning the safety aspects of the facility design, construction, operation, and plans for deactivation.

3.2 Control of the Authorization Basis

The AB documents for theWTP Project are considered configured items under Configuration Management. Changes to AB documents are managed by the WTP Project configuration management program.

3.3 Changes to the Authorization Basis

Changes to the authorization basis include changes to the facility design and administrative controls (e.g., procedures, programs, plans, or management processes) that are described in the authorization basis or are relied on to ensure conformance to the authorization basis. Changes to the authorization basis are managed by a configuration management program using the Project procedure for AB maintenance. All changes to the authorization basis will be in accordance with *Office of Safety Regulation Position on Contractor-Initiated Changes to the Authorization Basis* (RL/REG-97-13).

4.0 Internal Safety Oversight

Internal safety oversight for the WTP Project involves several oversight functions to ensure safety of the public and workers and to preclude environmental degradation. These internal safety oversight functions include corporate safety assessments, management assessments, independent assessments and audits, safety committees, incident investigations, maintenance of the authorization basis, and, during radiological operations, the USQ process. Assessments of the WTP Project verify that public and worker safety considerations are reflected in the design, procurement, construction, and commissioning of the facility. Assessments are covered in 24590-WTP-QAM-QA-01-001, *Quality Assurance Manual.* Several administrative functions provide information on the adequacy of the oversight functions and also provide information used to define the scope of future internal safety oversight functions. This information includes: performance monitoring; performance indicators; lessons learned and industry experience; and feedback and trending.

The following activities are part of internal safety oversight:

1) Conducting performance-based assessments that emphasize work activity in progress
2) Reporting deficient conditions to line management
3) Following up on corrective actions to prevent a recurrence of the deficiency
4) Applying performance trending to determine existence of programmatic issues and plan for future oversight areas
5) Understanding the requirements of the Price Anderson Amendments Act and 10 CFR 820, "Procedural Rules for DOE Nuclear Activities"
6) Assisting line management to establish a positive safety culture
7) Incorporating applicable lessons learned from previous WTP incidents and industry experience at other DOE sites and the commercial power industry relevant to the Project oversight program

8) Maintaining a continuing interaction with the WTP Project Regulator on the status and direction of project oversight activities.

Internal oversight may include participation of staff members external to the WTP Project Contractor. Members are selected based on their experience and qualifications to provide different perspectives or expertise in specific functional areas.

5.0 Integrated Safety Management

This chapter describes how safety management is integrated into work planning and performance. Lines of responsibility and authority for integrated safety management issues are described. Personnel qualification, resource allocation, and hazard assessments, controls, and operating conditions are discussed.

5.1 Integration Into Work Planning and Performance

The Project safety management process protects the public, workers, and the environment through implementing work practices that never compromise safety for the sake of production or expediency. This is achieved by way of the following:

1) Conduct activities in an atmosphere of trust and confidence based on open, honest, and responsible communication
2) Encourage employee feedback
3) Use proven and effective approaches to risk identification and control
4) Conduct business with integrity and mutual respect for employees and interfacing organizations
5) Apply a systematic approach to all activities that affect integrated safety management
6) Establish clear ownership and accountability
7) Define and reach agreement with the employees on the work to be accomplished by the facility operation and the expectation to accomplish the work in a safe manner
8) Promote teamwork through involvement of knowledgeable parties
9) Empower employees to effectively protect themselves, the public, and the environment
10) Allocate appropriate resources to support integrated safety management activities
11) Support continuous improvement of integrated safety management performance
12) Manage and conduct a consistent and project-wide integrated approach to integrated safety management for all activities
13) Encourage and promote sharing integrated safety management information and resources

Application of the above work practices allows the WTP Project team to effectively implement WTP Project Contractor guiding principles for integrating safety management into work planning and performance efforts. These guiding principles include establishing line management responsibility for integrated safety management, establishing and making clear lines of authority, ensuring that personnel have the necessary qualifications to perform the work, providing effective allocation of resources, performing pre-work hazard assessments, establishing appropriate controls for hazards and hazardous situations, and establishing operational requirements.

These work practices and principles are an integral part of the WTP Project team safety culture. They are formalized in WTP Project policies, procedures, and instructions and are incorporated into all activities described in the following sections. The flowdown of these work practices, and principles to subcontractors is discussed in Section 7.0 "Control of Subcontractors".

5.2 Line Management Responsibility for Integrated Safety Management

Line management responsibility and accountability for safety is one of the key principles of the WTP Contractor approach to safety management integration. To ensure maximum effectiveness in integrated safety management performance, employees are informed of their responsibility and accountability for creating and maintaining a safe and healthy workplace and protecting the environment.

In addition, safety management support individuals do not assume roles that reside with the line organization. This creates an environment where accountability is clearly focused and safety management priorities are never sacrificed to another line mission or objective.

5.3 Lines of Authority and Responsibility

Clear and unambiguous lines of authority and responsibility are established throughout the Project through its design, construction, operation, and deactivation phases. The flowdown of safety management responsibility and accountability starts with the WTP Project Manager and extends through the management and supervisory chain to each worker, irrespective of the type of work being performed. This flowdown is captured in policies and procedures, communicated to the workforce through orientation and training, reinforced by group and individual performance evaluations, and monitored and assessed by independent oversight provided by safety management professionals.

Stop-work authority also flows down from senior management to individual workers who are explicitly empowered to halt any activity in which they are engaged that is unsafe or potentially harmful to the environment.

6.0 Organization Roles, Responsibilities, and Authorities

The responsibility for the design, construction, commissioning, operation, and deactivation of the River Protection Project-Waste Treatment Plant lies with the designated WTP contractors throughout these various life-cycle phases of the WTP facility. These contractors to the Department of Energy, Office of River Protection will include the Design, Construction, and Commissioning (DC&C) contractor, the Operations contractor, and the Deactivation contractor.

These contractor's roles, responsibilities, and authorities include defining and implementing nuclear, radiological, and process safety standards and the related safety bases for protection of the WTP occupational workers and the public. These WTP contractors are responsible for defining and implementing DOE-approved safety standards and communicating those safety standards as requirements to all WTP Project team members and subcontractors who conduct work on the Project.

While the WTP Project team members manage subcontractors, the WTP contractors retain responsibility for oversight of team members and subcontractors performance and for overall project safety. The commitment inherent in this structure is that line management retains the responsibility for development and implementation of the safety basis. Although some specific roles may be reassigned within the organization, line management's responsibility for safety is invariant.

Overall Project roles, responsibilities, and authorities are provided in 24590-WTP-QAM-QA-01-001, *Quality Assurance Manual*. Project roles, responsibilities, and authorities related to radiological, nuclear, and process safety for the DC&C contractor shall be clearly defined. Envisioned roles, responsibilities, and authorities related to radiological, nuclear, and process safety for the Operations contractor shall be clearly defined.

7.0 Control of Subcontractors

The WTP Contractor is responsible for ensuring that all subcontractors work as safely as the WTP Project employees. The WTP Project Contractor's responsibilities include the following:

1) Informing the subcontractors of known fire, explosion, or toxic hazards relating to the subcontractor's work and the process
2) Explaining to the subcontractor the applicable provisions of the emergency plan
3) Developing and implementing safe work practices to control the entrance, presence, and exit of subcontractor employees, including their presence in areas of the process covered by the PSM standard
4) Periodically evaluating the performance of subcontractors in fulfilling their obligations as stated
5) Maintaining an illness and injury log relating to the subcontractor work in the process areas

Each subcontractor's responsibilities include the following:

1) Ensuring that subcontractor employees are trained in the work practices necessary to safely perform their assignments
2) Ensuring that subcontractor employees are instructed in the known hazards of the process as related to their job assignments, and in the relevant provisions of the emergency management plan
3) Documenting that each subcontractor employee has received and understood the training required to work safely at the WTP
4) Ensuring that each subcontractor employee follow the safety rules of the WTP and the site safe work practices, and advise the contractor of any unique hazards presented or found during the course of the subcontractor's work

Project integrated safety management requirements are imposed on subcontractors in contracting documents. Subcontractors are required to appoint an integrated safety management representative who is the interface with the WTP Project team on all integrated safety management matters.

To ensure that WTP subcontractors are performing their work safely, both formal and informal safety reviews and assessments are performed. Results of these evaluations are transmitted to both Project management and to the affected subcontractors.

Appendix J

Ad Hoc Implementing Standard for Startup

CONTENTS

1.0 Introduction

The Department of Energy (DOE) regulatory approach requires that the Contractor take an active role in identifying and recommending the standards and requirements it will use to achieve adequate safety for its specific activities. Through regulation DOE has specified that the integrated safety management process be used to identify the subordinate standards that will be used to conform to the top-level safety standards and principles specified by the DOE [ref. 1].

The DOE has specified top-level standards for testing of the Waste Treatment Project (WTP) [ref. 2]. This subordinate standard has been developed to ensure the top-level standards related to testing will be properly implemented.

2.0 Scope

This subordinate standard defines the testing requirements of WTP important to safety (ITS) systems, structures, and components (SSCs). It contains a discussion of non-ITS SSCs to define the interface between ITS and non-ITS SSCs. The testing requirements of non-ITS SSCs are not regulated by this standard.

3.0 Definitions

- Component Test:

  Test performed on designated components.
- Acceptance Test:

  System-level test performed on systems designated as Safety Design Class (SDC) or Safety Design Significant (SDS).
- Functional Test:

  System-level test performed on systems designated as Risk Reduction Class, or system not designated as important to safety.
- Integrated Water Run:

  Test performed at the facility level using water as the process fluid. Integrated Water Runs are performed on the Pretreatment, Low Activity Waste, and High Level Waste facilities.
- Cold Commissioning Test:

  Test performed at the facility level using process reagents and non-radioactive simulants as the process fluid. Cold Commissioning tests are performed on the Pretreatment, Low Activity Waste, and High Level Waste facilities.
- Hot Commissioning Test:

  Test performed at the facility level using radioactive wastes as the process fluid.

4.0 Test Phases

The WTP test program shall use a phased approach to testing. The test phases are:

- Component Testing
- System Testing (Acceptance Tests and Functional Tests)
- Integrated Water Runs
- Cold Commissioning Testing
- Hot Commissioning Testing

5.0 Graded Approach

The WTP test program shall be developed using a graded approach to ensure the greatest attention is given to the most important systems, structure, or components (SSCs).

All WTP SSCs are classified as either important to safety (ITS) or not important to safety (non-ITS). The WTP has further classified SSCs into a three-tiered ITS classification system:

- Safety Design Class (SDC)
- Safety Design Significant (SDS)
- Risk Reduction Class (RRC)

Safety Requirements Document, Volume II, Safety Criterion 1.0-6 contains the attributes for each classification [Ref 3]. Testing of designated components shall be performed either by a vendor or by the test organization. Component test procedures and test results shall be reviewed and approved in accordance with the test program administrative procedures.

Testing of SDC or SDS systems shall be performed and documented using Acceptance Tests. Acceptance Test procedures and their test results shall be reviewed and approved by the Joint Test Group. The Chairman of the Joint Test Group shall be the Area Test Manager, or his designee. The test program administrative procedures shall define the Joint Test Group members and their responsibilities.

Testing of RRC systems shall be performed and documented using Functional Tests. Functional Test procedures and their test results shall be reviewed and approved in accordance with the test program administrative procedures.

Testing of non-ITS systems shall be performed and documented using Functional Tests. Functional Test procedures and their test results shall be reviewed and approved in accordance the test program administrative procedures. Testing may not be required on designated non-ITS systems, such as lighting and lightning protection. Acceptance tests performed on SDC and SDS systems and Functional tests performed on RRC systems will demonstrate that the systems have been properly constructed and can perform their safety functions.

Integrated Water Runs Test procedures and their test results shall be reviewed and approved by the Joint Test Group.

Cold Commissioning and Hot Commissioning test procedures and their test results shall be reviewed and approved by the Commissioning Review Board. The Commissioning Review board shall be chaired by the Contractor with Department of Energy participation as defined in WTP test program administrative procedures.

Table 5-1 provides a summary of WTP test requirements.

Table 5-1 Test Description Summary

Category	Procedure Type	Performed By	Procedure Review and Approval	Test Results Approval
Individual Component	Component Test	Vendor or test staff	In accordance with test program administrative procedures	In accordance with test program administrative procedures
SDC or SDS System	Acceptance Test	test staff	Joint Test Group	Joint Test Group
RRC System	Functional Test	test staff	In accordance with test program administrative procedures	In accordance with test program administrative procedures
Non-ITS System[1]	Functional Test	test staff	In accordance with test program administrative procedures	In accordance with test program administrative procedures
Integrated Test Using Water as Process Fluid	Integrated Water Run Test	test staff	Joint Test Group	Joint Test Group
Integrated Test Using Simulant as Process Fluid	Cold Commissioning Test	test staff	Commissioning Review Board	Commissioning Review Board
Integrated Test Using Radioactive Waste as Process Fluid	Hot Commissioning Test	test staff	Commissioning Review Board	Commissioning Review Board

Note 1: Testing may not be required on designated non-ITS systems, as approved by Area Test Manager in accordance with test program administrative procedures.

6.0 Construction Turnover

The WTP test program and construction administrative procedures shall define the process and controls required for system turnover from the construction organization to the test organization. The procedures shall include instructions for:

- Pre-turnover system review between the construction and test organizations
- Identification of activities to be reviewed during system turnover
- Submittal of as-built* documentation for the system

- Identification, tracking and resolution of open items
- System turnover approval process

*As built in this context refers to documentation of current system configuration

7.0 Test Procedures

The WTP test program administrative procedures shall define the required content of test procedures. The following types of procedures shall be addressed:

- Component test procedures
- System test procedures (Acceptance and Functional Test)
- Integrated water run test procedures
- Cold commissioning test procedures
- Hot commissioning test procedures

The test program administrative procedures shall provide guidance for the following test procedural elements:

- Format
- Review and approval process
- Test acceptance criteria
- Evaluation of open items prior to performing Acceptance or Functional Tests
- Reporting, and resolution of test deficiencies
- Review and approval of test results
- Recording of baseline data

8.0 Validation of Operating and Maintenance Procedures

Operations procedures for the WTP will be drafted, reviewed, verified, validated, and approved per the WTP Conduct of Operations Program. Validated procedures will be provided to the testing organization for use during initial system startup and other testing activities as needed. Approved operating and maintenance procedures shall be performed as required during the period between system turnover from Construction and hot commissioning testing. Procedural inadequacies discovered during this testing period will be corrected in accordance with project administrative procedures. The approval of the operating and maintenance procedures before their performance will ensure that the procedure is compatible with the equipment or system being maintained, and that it provides sufficient and understandable guidance to the end user.

9.0 Test Acceptance Criteria

The Engineering organization shall develop a plan that will define its responsibility in defining acceptance criteria for the WTP test program. Test acceptance criteria of ITS components or systems shall consider the uncertainties used in accident analysis, if applicable.

10.0 Retest

The WTP test program administrative procedures shall establish controls to ensure adequate retest of system functions repaired or modified after completion of component tests, Acceptance Tests, Functional Tests, Integrated Water Run Tests, Cold Commissioning Tests, or Hot Commissioning Tests. The administrative controls shall consider a graded approach to testing so that the required retest is commensurate with the controls of the initial test.

11.0 Readiness Assessments

A readiness self-assessment shall be required prior to entry into cold commissioning testing (after the completion of integrated water runs). A readiness assessment shall be conducted prior to entry into hot commissioning testing (at the completion of cold commissioning testing). The results of these assessments will be submitted to DOE for evaluation and in support of authorization decisions and regulatory oversight. A process safety hazard analysis will be performed and recommendations will be resolved or implemented prior to entry into cold commissioning testing.

The cold commissioning readiness self-assessment shall confirm the following:

- Testing Acceptance Tests and Functional Tests have demonstrated that the facility is prepared to support cold commissioning.
- The facility is staffed with a sufficient number of trained and qualified staff to support cold commissioning.
- Facility procedures necessary to support cold commissioning are approved and ready for use.

The hot commissioning readiness assessment shall be performed by personnel independent of the operating and testing staffs. The review shall evaluate the following areas:

- Safety Documentation
- Hardware and Systems
- Personnel
- Programs and Procedures
- Regulatory Compliance

The Project Safety Committee shall approve the results of this review before the Waste Treatment Plant certifies to the US Department of Energy Office of River Protection that the project is ready for operation with active waste.

12.0 Schedule

The WTP project shall maintain a schedule that identifies the following activities:

- System turnover from the construction organization to the test organization
- Acceptance and Functional Test preparation
- Performance of Acceptance Tests and Functional Tests
- Integrated Water Run Tests
- Cold Commissioning Tests
- Hot Commissioning Tests

The development and maintenance of this schedule is not an ITS activity.

13.0 Records

Test records shall be maintained to satisfy the requirements of the Quality Assurance Manual, Policy Q-17-1, "Quality Assurance Records".

14.0 References

1 DOE/RL-96-0004, Revision 2, *Process for Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for the RPP Waste Treatment Plant Contractor.*

2 DOE/RL-96-0006, Revision 2, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for the RPP Waste Treatment Plant Contractor.*

3 24590-WTP-SRD-ESH-01-001-02, Rev 1h, *Safety Requirements Document Volume II.*

4 24590-WTP-QAM-QA-01-001, *Quality Assurance Manual.*

Appendix K

Facility Areas Not Requiring Automatic Fire Suppression Systems Based on High Radiation and Low Combustible Loading

K.1: List of HLW Facility Areas Not Requiring Automatic Fire Suppression Systems Based on High Radiation and Low Combustible Loading.

HLW Area	Description	PFHA[a] Combustible Loading[b]
H-136	Canister Handling Cave	Very Low
H-B015	Drum Transfer Tunnel	Very Low
H-B035	Canister Decon Cave	Low
H-B014	Wet Process Cell	Very Low
H-B032	Pour Tunnel No. 1	Very Low
H-B005A	Pour Tunnel No. 2	Very Low
H-B021	SBS Drain Collection Cell No. 1	Very Low
H-B005	SBS Drain Collection Cell No. 2	Very Low
H-B013	Active Pipeway to/from Pretreatment	Very Low

a Preliminary Fire Hazard Analysis

b "Very Low" means an average combustible load, CL< 20,000 Btu/ft^2 with isolated concentrations ≤ 40,000 Btu/ft^2 - "Low" means an average combustible load, 20,000 Btu/ft^2 ≤ CL ≤ 80, 000 Btu/ft^2 with isolated concentrations ≤ 160,000 Btu/ft^2

K.2: Criteria for the Omission of Automatic Fire Suppression Systems in Waste Treatment and Immobilization Plant (WTP) Process Buildings

Automatic fire suppression systems (e.g., sprinklers), including fire detection in lieu of sprinklers as required by the International Building Code (IBC) and other SRD implementing codes and standards, may be omitted in any WTP space provided **all** of the following criteria are met:

Combustible Loading

1. Average equivalent combustible loading is 2000 BTU/square foot or less.
2. Isolated concentrations of combustibles do not exceed 160,000 BTU.
   - If fully enclosed or shielded by noncombustible material, concentrations of combustibles are separated by at least four (4) feet.
   - If exposed, concentrations of combustibles are separated by at least ten (10) feet from each other and from combustible surfaces.

Accessibility

1. Access is not available through doors or hatches or other permanent means of personnel entry.
2. The space is continuously R5 (i.e., extraordinary steps would be required to reduce radiation levels below R5 for personnel entry). As used here, R5 includes both High and Very High Radiation Areas (i.e., areas with a radiation field strength equivalent to 100 mrem or higher per hour at 30 centimeters).

K.3: WTP Process Buildings, by Room, Receiving DOE Approval for Omission of Automatic Fire Suppression Systems

Low Activity Waste Rooms Approved to Omit Automatic Fire Suppression Systems:

Room	Description
L-B025B	Container Transfer Corridor
L-B025C	Container Buffer Store
L-B025D	Container Buffer Store
L-B011C	Pour Cave
L-B013B	Pour Cave
L-B013C	Pour Cave
L-B015A	Pour Cave
L-123	Wet Process Cell
L-124	Wet Process Cell
L-126	Effluent Cell

High Level Waste Rooms Approved to Omit Automatic Fire Suppression Systems

Room	Description
H-B039B	Canister Rinse Tunnel
H-104	Filter Cave
HP-104A	Filter Cave Platform
H-132	Canister Storage Cave
H-117	Melter Cave #1
H-106	Melter Cave #2

Pretreatment Facility Rooms Approved to Omit Automatic Fire Suppression Systems

Room	Description
P-123	Hot Cell
P-123A	Remote Decon Maint Cave
P-335	Filter Cave
P-335A	Filter Cave Decon Chamber

Appendix L

Ad Hoc Implementing Standard for Seismic Design of Pressure Vessels

1 Design Codes and Requirements

The pressure vessel design code for WTP is *ASME Boiler and Pressure Vessel Code*, Section VIII, "Rules for Construction of Pressure Vessels". However, the WTP is also required to meet the DOE seismic requirements specified in DOE-STD-1020-94, *Natural Phenomena Hazards and Evaluation for Department of Energy Facilities* as this standard is tailored for the WTP in Appendix C. ASME Section VIII requires that the loadings to be considered in designing a vessel shall include those from seismic reactions where required. ASME Section VIII requires that for the combination of earthquake loading with other loadings, the wall thickness of a vessel computed by these rules shall be determined such that the general membrane stress shall not exceed 1.2 times the maximum allowable stress values used for normal loadings. These allowable stresses will be applied to the vessel.

ASME Section VIII, Division 1, provides the basic design principles and formulas for the design of pressure vessels. ASME Section VIII contains mandatory guidance for pressure vessel materials, design, fabrication, examination, inspection, testing, certification, and pressure relief. ASME Section VIII, Division 1, requires that seismic reactions be considered in designing a vessel. However, it does not specify how seismic loads are to be considered. It does not contain rules to cover all details of design and construction. Where complete details are not given, it is intended that the designer shall provide details of design and construction that will be as safe as those provided by the rules of the ASME Section VIII. ASME Section VIII, Division 2, Appendix 4, provides a methodology for performing stress analyses on vessels. However, in order to ensure that the stresses in the vessel comply with the requirements of ASME Section VIII, Division 1, the acceptance criteria for SC-I, SC-II, SC-III, and SC-IV vessels shall be in accordance with Appendix 4 of ASME Section VIII, Division 2, using the allowable stress, S, from ASME Section VIII, Division 1, in lieu of the design stress intensity, S_m, of ASME Section VIII, Division 2.

The details of the vessel supports supplied with the vessel, such as skirts and saddles, will conform to good structural practice in accordance with the AISC *Manual for Steel Construction* as recommended by ASME Section VIII, Appendix G. AISC *Manual for Steel Construction - Allowable Stress Design,* Ninth Edition, will be used.

The weld of vessels to the embedded structure will be in accordance with the AISC *Manual for Steel Construction* as specified by the vessel vendor. The embedded structural supports for the vessels and the bolts, studs and nuts securing the vessels to the embeds are designed in accordance with the applicable requirements of ACI 318-99, *Building Code Requirements for Structural Concrete,* ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*, ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities,* or UBC-97, *Uniform Building Code.*

The governing design code for the vessel proper is ASME Section VIII, Division 1.

The governing design code for the vessel supports supplied with the vessel proper is the AISC *Manual for Steel Construction* per paragraph UG-54 and Appendix G of the ASME Section VIII, Division 1. AISC *Manual for Steel Construction - Allowable Stress Design,* Ninth Edition, will be used.

The governing code for the weld of the vessel to the embedded structures shall be in accordance with the AISC *Manual for Steel Construction*. The governing design codes for the embedded structure for the vessels and the bolts, studs and nuts securing the vessel to the embeds shall be in accordance with applicable requirements of ACI 318-99, *Building Code Requirements for Structural Concrete*, ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*, ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*, or UBC-97, *Uniform Building Code*.

Internal components, supports and piping systems shall be analyzed the same as the parent vessel unless otherwise noted.

2 Load Combinations for Pressure Vessels

According to ASME Section VIII, in addition to loadings caused by internal or external design pressure, weight of the vessel and normal contents under operating or test conditions, superimposed static reactions, attachments, cyclic and dynamic reactions, impact reactions, temperature gradients and thermal expansions, and abnormal pressures, the pressure vessel must be designed for loads caused by wind, snow, and seismic reactions. Earthquake loading and wind loading need not be considered to act simultaneously.

The loadings to be considered in designing the vessel shall include those listed in paragraph UG-22 of ASME Section VIII, Division 1.

2.1 Seismic Category I and Seismic Category II Loads

The seismic analysis of SC-I and SC-II vessels and their supports shall be by the dynamic analysis method. The dynamic analysis shall be accomplished using the response spectrum, frequency domain, or time history approach. The seismic loads shall be considered acting simultaneously in three directions. A finite element model, which includes the mass of the contained liquid shall be used, or procedures described in Section 3.5.4 of ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary* or Chapter 4 of BNL 52361, *Seismic Design and Evaluation Guidelines for the Department of Energy High-Level Storage Tanks and Appurtenances*, shall be followed.

2.2 Seismic Category III and IV Loads

The seismic loads for SC-III and SC-IV vessels and their supports shall be in accordance with the UBC-97, *Uniform Building Code*.

3 Allowable Stresses

3.1 Maximum Allowable Tensile Stress

The maximum allowable tensile stress, S, for the material of construction of the vessel shall be as specified in ASME Section II, Part D, Subpart 1.

3.2 Maximum Allowable Longitudinal Compressive Stress

The maximum allowable longitudinal compressive stress used in the vessel design shall meet the requirements of paragraph UG-23 (b) of the ASME Section VIII, Division 1.

3.3 Maximum General Primary Membrane Stress

The wall thickness of a vessel shall be determined such that the induced maximum general primary membrane stress does not exceed the maximum allowable stress in tension for any combination of loadings listed in paragraph UG-22 of ASME Section VIII, Division 1, that induce primary stresses and are expected to occur simultaneously during normal operation of the vessel.

3.4 Combined Primary Membrane Plus Primary Bending Stress

The combination of loadings listed in paragraph UG-22 of ASME Section VIII, Division 1 shall not induce a combined maximum primary membrane stress plus primary bending stress across the vessel wall thickness, that exceeds 1.5 times the maximum allowable stress value in tension.

3.5 Combination of Seismic Loadings with Other Loadings

For the combination of seismic loading with other loadings per UG-22, the wall thickness of a vessel shall be determined such that the general primary membrane stress shall not exceed 1.2 times the permitted maximum allowable stress specified in Sections 3.1, 3.2, 3.3, or 3.4 above. Seismic loading and wind loading need not be considered to act simultaneously.

3.6 Stress Analysis Performed in Accordance with ASME Section VIII, Division 2, Appendix 4

The acceptance criteria for SC-I, SC-II, SC-III, and SC-IV vessels shall be in accordance with Appendix 4 of ASME Section VIII, Division 2, using the allowable stress, S, from ASME Section VIII, Division 1 in lieu of the design stress intensity, S_m, of ASME Section VIII, Division 2.

3.7 Maximum Allowable Stresses and Acceptance Criteria for Vessel Supports

Detailed design of vessel supports shall be in accordance with the recommendation of ASME Section VIII, Division 1, Appendix G. The stresses in vessel supports shall not exceed the maximum allowable stress values for the material of construction per Part 5 of the AISC *Manual for Steel Construction*.

3.8 Acceptance Criteria for Structural Ring Supports and Securing Welds, Bolts, Studs and Nuts

The acceptance criteria for the weld of vessels to the embedded structure will be in accordance with the AISC *Manual for Steel Construction* as specified by the vessel vendor. The acceptance criteria for the structural ring supports and bolts, studs and nuts securing the vessels to the structural ring supports shall be in accordance with applicable requirements of ACI 318-99, *Building Code Requirements for Structural Concrete,* ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*, ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities,* or UBC-97, *Uniform Building Code.*