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Overview and Summary of NRC Involvement with DOE in the Tank Waste Remediation System-Privatization (TWRS-P) Program

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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Overview and Summary of NRC Involvement with DOE in the Tank Waste Remediation System-Privatization (TWRS-P) Program

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ABSTRACT

In 1995, the U.S. Department of Energy (DOE) embarked on an effort to privatize the processing through vitrification of 54 million gallons of radioactive waste that has been stored in 177 underground storage tanks at the Hanford Site. The U.S. Nuclear Regulatory Commission (NRC) provided assistance to DOE on the Tank Waste Remediation System-Privatization (TWRS-P) program, with a potential transition to NRC regulatory authority at a future time. In 2000, DOE terminated the privatization approach, and decided to use more traditional contracting methods.

During their reviews, NRC staff analyzed both unmitigated and mitigated consequences from potential accident scenarios at the proposed facility. NRC staff's efforts identified several key areas of uncertainty, such as melter failure modes and frequencies, that would require further study before more refined analyses could be performed. The reviews also identified several open issues, including the need for significantly more detailed design information and safety analyses, and greater defense-in-depth. In particular, the design, at the time of termination of the privatization, was found to be very preliminary and corresponded to perhaps a 15 percent level of design.

This report summarizes NRC's participation in and observations on the TWRS-P program and identifies issues from the NRC's perspective.

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THE TANK WASTE REMEDIATION SYSTEM-PRIVATIZATION PROGRAM

In 1995, the U.S. Department of Energy (DOE) embarked on an effort to privatize the processing through vitrification of 54 million gallons of radioactive waste that has been stored in 177 underground storage tanks at the Hanford Site. Under the initial phase of the Privatization plan, fixed-price waste treatment services for processing a portion of the waste was to be supplied, on leased land at the Hanford Site, by contractor-owned, contractor-operated facilities under a fixed-priced contract.

DOE established a dedicated Regulatory Unit (RU) led by a Regulatory Official (RO) at the DOE Richland Operations Office with regulatory authority exclusive to the regulation of TWRS-P contractors. The RO reported directly to the Manager of DOE/Richland Operations Office (RL) at a level equivalent to the DOE Program Manager for TWRS. The RU planned on following the five principles of good regulation as articulated by the NRC - independence, openness, efficiency, clarity, and reliability. The basic concept of DOE's regulatory approach at TWRS-P was that the contractor is responsible for achieving adequate safety, complying with applicable laws and regulations, and conforming with top-level safety standards and principles stipulated by DOE. Consistent with applicable laws and regulations, the contractor is required to tailor the exercise of this responsibility to the specific hazards associated with its activities, and is encouraged to do this in a cost-effective manner that applies best commercial practices. TWRS-P contractors have the responsibility to identify and recommend to DOE the set of standards, regulations, and requirements necessary to ensure adequate safety at the proposed facilities. This constitutes a risk-based, integrated safety management (ISM) process. DOE's responsibility is to execute the regulatory process, including authorization of contractor actions and confirmation that the contractor activities are performed safely and within approved limits. The authority of the RU to regulate a TWRS-P contractor is derived from the terms of the TWRS-P contract ("regulate by the contract").

The U.S. Nuclear Regulatory Commission (NRC) provided assistance to DOE on the TWRS-P program for 3 ½ years under a Memorandum of Understanding (MOU) signed in January 1997. The MOU provided for NRC to acquire an understanding of the wastes and potential treatment processes, assist the DOE in performing reviews in a manner consistent with the NRC's regulatory approach for commercial nuclear facilities, and develop an effective regulatory program for the potential transition to NRC regulatory authority at a future time. In May 2000, DOE abandoned the privatization approach for cost reasons, and declared its intent to pursue a more conventional, management and operations (M&O) style contract for the design, construction, and operation of the waste treatment facilities. The M&O contractor may or may not use the designs, technologies, and approaches already developed. With this contract change, DOE also signaled its intent to self-regulate the facilities for the foreseeable future, without any schedule for transition to NRC regulatory authority. As a result of these changes, the NRC has re-evaluated its role in the program. The NRC has determined that DOE's decision to terminate the privatization contractor and pursue the M&O approach effectively completes the MOU by its own terms. Consequently, NRC participation has ceased. However, the NRC remains willing to discuss possible arrangements with DOE for NRC involvement in the Hanford tank waste programs in the future, under a new MOU, if there is again a need for NRC expertise and if the NRC sees benefit in such involvement, such as the potential transition to NRC regulatory authority.

This report summarizes NRC's participation in and observations on the TWRS-P program, identifies potential open issues from the perspective of the NRC staff, and notes concerns with deviations from the NRC's regulatory approach. Unless otherwise noted, this report is based upon the design as it existed in June 2000.

Section 1 provides more information on the program.

NRC PARTICIPATION

When NRC began its involvement, the TWRS-P program was initially designed to begin with a relatively small pilot plant approach and facility for early processing of the wastes. Such an approach would have allowed verification of design and technical approaches with minimal economic, programmatic, and safety risks, and would still have resulted in the processing of some of the waste materials. However, due to programmatic changes, including concerns regarding the feasibility of privately financing a short-term facility, DOE decided to pursue a much larger, full-scale facility instead of a pilot plant. This decision greatly increased the flow rates and radiochemical inventories for the proposed facility and contributed to several design basis/authorization issues encountered during the program.

In carrying out its responsibilities under the MOU, NRC staff participated with DOE in technical reviews and meetings of various contractor submittals including, for example, Safety Requirements Documents (SRDs), Hazard Analysis Reports (HARs), Initial Safety Analysis Reports (ISARs), Design Safety Features (DSF) submittals, and the Firm Fixed Price (FFP) submittal. NRC staff also reviewed numerous other documents on specific features and concerns (e.g., seismic design, quality assurance/quality control (QA/QC), radiological plans, fire protection, chemical safety, etc.) and attended many safety and regulatory meetings with DOE and DOE's contractors (e.g., monthly Topical Meetings, design review meetings, etc.). Oral and written comments were provided by NRC staff to DOE as a result of these reviews and participation in these meetings. NRC staff also assisted DOE in the development of appropriate regulatory guidance and the NRC issued a final Standard Review Plan (SRP) for TWRS-P facilities for use in any future NRC regulatory oversight of the TWRS-P facilities. While participating in this program, NRC staff became fully cognizant of the waste issues, design requirements, safety, and regulation of the proposed facility, thus meeting the NRC's primary objectives of the MOU.

To supplement their reviews of contractor and DOE documentation, and to gain better insights into the environmental, safety, and health (ES&H) characteristics of the proposed TWRS-P facilities, NRC staff analyzed both unmitigated and mitigated consequences from potential accident scenarios at a generic TWRS-P facility with similar characteristics and operations to those proposed by the TWRS-P contractors. The analyses used a conservative approach considered to be suitable for safety categorizations and preliminary designs. Several scenarios involving large radiochemical inventories (in tanks), flammable gases, organic ion exchange resin interactions, glass melters, and cold chemical effects were found to have potential accident consequences to the workers and the public of significant severity and risk (1E-2/yr to 1E-4/yr). Under such circumstances, the NRC staff concluded that accident prevention (reduced probability) and mitigation (reduced consequences) would become necessary, requiring the identification of improved design approaches and items relied on for safety. Ideally, processes and approaches proposed for tank waste processing would incorporate robust designs with redundant features. The NRC staff noted that suitable process accident prevention and mitigation methods exist that are compatible with the regulations and offer the potential for reducing process accident risk to more acceptable levels (circa 2E-6/yr). Furthermore, the NRC staff found that relatively standard nuclear industry methods (e.g., high efficiency particulate air filter systems) could be used for risk reduction; with the possible exception of the melter areas, no unique or new risk reduction methods that might require qualification appeared to be necessary. NRC staff's efforts identified several key areas of uncertainty, such as melter failure modes and frequencies, unique design features, and corrosion resistant requirements, that would require further study before more refined analyses could be performed. The insights gained from these efforts allowed the NRC staff to better understand the potential safety issues and risk control strategies associated with the TWRS-P program.

Chapters 2.0 and 3.0 of this report provide more information on the NRC involvement in TWRS-P and NRC assessment of several areas of review.

POTENTIAL OPEN ISSUES

As a result of the NRC staff's technical review of documentation and participation in meetings with DOE and the contractors, several concerns and potential open issues were identified. These include the need for significantly more detailed design information and safety analyses, and greater defense in depth. In particular, the design at the time of termination of the privatization contract was found to be very preliminary and corresponded to perhaps a 15 percent level of design. The NRC staff has identified over two-dozen significant issues and over fifty specific topics in the current design and approach that would require further efforts and analysis to achieve adequate closure; these issues are discussed further in Chapter 3.0 of this report and in Appendix A. These significant issues include both programmatic aspects of TWRS-P (e.g., maintenance of design/authorization basis, level of detail) and technical issues (e.g., large volumes of tankage and radionuclide inventories, combined chemical and radiological hazards, melter corrosion). DOE, as the current regulator, has also identified similar issues.

The melters present several issues, due to their size, capacities, and surface area fluxes, all of which would become the largest for radwaste vitrification in the world. However, the experiential base, particularly from the perspective of potential ES&H concerns, is limited. Towards the end of the program, the Contractor identified the need for high nickel alloys for corrosion resistance in areas of the melter that would usually be made of more conventional materials (e.g., carbon steel) in existing vitrification facilities. This was based upon testing a one-third scale melter, but no further information was made available. The Contractor also presented analyses that implied a relatively high level of risk to the worker (circa 1E-3/yr) from a melter offgas/NOx scenario. The melter designs also have several unique attributes, including a thin gap between the cooling coils and the outer steel casing, and drainage holes. More information and analyses would be required to ascertain the safety ramifications if these melter designs are used by the new contractors.

DOE prescribed an expedited schedule at the beginning of the program, with limited flexibility. Consequently, throughout the length of the program, the NRC and DOE staff technical reviews were held to tight schedules (typically a 2 week turnaround for a multivolume submittal) which frequently resulted in the inability to identify action items and plans, and achieve full closure on a number of the issues. Consequently, resolution of several significant design and safety issues (such as those discussed in Chapter 3.0 and Appendix A of this report) may not occur for some time. In addition, the likely impacts from further contractor changes are unclear but would likely imply more uncertainties and more design changes that, in turn, could raise more issues and the corresponding need for additional time for review and resolution prior to proceeding into construction and operation.

POTENTIAL CONCERNS WITH DIFFERENCES FROM THE NRC'S REGULATORY APPROACH

The working relationship between the NRC and the DOE has evolved during the program, and DOE has acknowledged the value added by the NRC's involvement. In the opinion of the NRC staff, there are several significant concerns which appear to be having a deleterious effect upon DOE's regulatory approach. These are discussed further in Chapter 4.0 of this report. The most notable of these are summarized as follows:

1. <u>The influence of programmatic issues (including cost, schedule, and capacity) upon the regulatory review activities:</u> Programmatic issues, including economics, arose in several regulatory and safety-related meetings, usually in conjunction with the discussions regarding safety-related components and systems, and defense-in-depth. Short schedules (typically 2 weeks) were established by DOE for reviewing large, multivolume submittals and may have impacted the depth and quality of the reviews, including the identification of safety and regulatory concerns. (DOE-Headquarters also expressed concerns about schedule pressures - see page 28.) In addition, DOE's programmatic concerns and desires emphasized higher throughputs (potentially a four-fold or larger increase) and/or additional/larger facilities to increase waste processing rates and reduce schedules and costs. Many design changes were made to accommodate this emphasis on cost and schedule. However, the safety analyses do not appear to have reviewed the potential impacts from such higher throughputs or additional/larger facilities. The emphasis also appeared to contribute to the deferral of some issues to subsequent reviews in order to maintain the schedule. Thus, the design work often continued with the potential for less than adequate consideration and closure of regulatory issues, such as ALARA (as low as reasonably achievable) (e.g., see Chapter 3.0 and Appendix A of this report).

2. <u>Maintenance of design/authorization basis (license):</u> Throughout most of the design effort (about 2 years), the design and safety teams of the Contractor (BNFL Inc.) worked quasi-independently, partly because of the previously mentioned emphasis on schedule. This led to inconsistencies between the design and the safety documentation. Changes in fundamental aspects of the design occurred in this time period, potentially without adequate consideration of regulatory needs, such as inventories and source terms for the

safety analyses. The design and authorization basis documents were not updated and few amendment requests were submitted to the RU. Ultimately, the RU delivered a Corrective Action Notice (CAN) on this concern in March 2000, which, in turn, led to a Corrective Action Plan (CAP) that was agreed to between the RU and the Contractor. A significant number of amendment requests were received by the RU from the Contractor after the CAN. However, by this time, most of the preliminary design activities in this phase of the program were completed and this timing of the CAN provided little room for regulatory review of the modifications and regulatory impact upon the design and design activities. Ironically, the flexible regulatory framework may have contributed to this situation; DOE postponed inspections for about a year that could have identified this situation and that could have potentially endorsed a CAP for correcting the situation.

3. <u>The application of a risk-based approach to the development of the design without additional considerations:</u> Risk-based analyses were used as the basis for the ISM process, which includes hazards identification, consequence estimation, and control mitigation. This is essentially a completely fluid process without a minimum level of requirements and, as practiced on TWRS-P, did not appear to adequately address unknowns, uncertainties, errors, proven practices, future plans, and experience. The ISM approach applied to TWRS-P resulted in less conservatism, reduced margins, and less defense-in-depth during the preliminary design phase - a phase when margins and conservatism would normally be relatively larger. There appeared to be more emphasis on the process, and less on the results. The Integrated Safety Management Plan (ISMP) is an iterative process which includes reassessing assumptions (e.g., of radionuclide concentrations) that can reduce a hazard below a limit into a bin with less reliability and safety requirements. Once this occurs, the scenario represents less risk with fewer safety requirements, and, again, since ISMP focuses on events with higher risk, this scenario and the underlying assumptions may be subject to less scrutiny. An unintended consequence of ISM at TWRS-P accrued from this circular logic: ISM focused on the higher risk areas by challenging assumptions; assumptions that were changed and resulted in lower consequences and risks may not have been revisited or re-evaluated as thoroughly. Thus, reduced consequence estimates may have resulted, and control strategies and equipment may not have been adequately identified. In contrast, the NRC regulatory approach applies a risk-informed, performance-based approach with defense-in-depth, appropriate levels of conservatism, and a minimum set of standards and requirements that are codified in the regulations.

4. <u>Limited use of NRC regulations and guidance:</u> DOE has adopted the NRC principles of good regulation in the documents that form the basis for the TWRS-P regulatory approach. However, DOE has not adopted the use of NRC regulations and guidance for TWRS-P, such as 10 CFR Part 70 and the TWRS Standard Review Plan. Instead, standards, codes, and regulations were selected by the Contractor and approved for their use by the RU with the application of ISM.

REGULATORY TRANSITION ISSUES

The NRC and the DOE have previously discussed issues related to the potential regulatory transition of TWRS-P to NRC regulation in the near future. Many of these issues are summarized in Chapter 5.0 and discussed in detail in Appendix D of this report, and have been discussed between the DOE and NRC over the 3-year length of the program. DOE is converting the contracts to an M&O arrangement for TWRS, which has been renamed the RPP/WTP (River Protection Project/ Waste Treatment Plant). The NRC staff believes that enabling legislation from Congress is desirable for the NRC to regulate either a privatized TWRS-P facility or an RPP/WTP and that any resulting issues are resolvable. From the viewpoint of the NRC staff, most of the regulatory issues would be addressed by the legislation that enables NRC regulatory authority over the TWRS/WTP facilities or NRC external regulation of DOE facilities, and by continued refinement and detailing of the proposed facility designs. The remaining issues relate to DOE programmatic activities and not regulation.

THE FUTURE TWRS/WTP PROGRAM

As previously noted, DOE has terminated the current privatization contract and approach and elected to follow an M&O contracting approach. According to the Request for Proposals (RFP), DOE plans to regulate these TWRS facilities. The specific features of the regulatory approach and the balancing of programmatic and safety issues are not identified as of this writing (December 2000). The means to follow, address, and close the design, safety, and regulatory issues identified from the NRC reviews and summarized in this report also have not been presented at this time. The RFP does include significant incentives for a contractor to reduce costs but does not mention safety as an evaluation factor.

The DOE has acknowledged the value added to the program by the NRC's participation. However, the NRC has terminated its involvement with the TWRS Project and has deployed staff to other projects. The NRC has also informed DOE that it is willing to discuss possible arrangements for NRC involvement in the future, if there is again a need for NRC expertise and if the NRC sees a benefit from such involvement, such as the potential transition to NRC regulatory authority

ABBREVIATIONS AND ACRONYMS

ACGIH	American Council of Government and Industrial Hygienists
ACNW	Advisory Committee on Nuclear Waste
ACRS	Advisory Committee on Reactor Safeguards
AEA	Atomic Energy Act
AIChE	American Institute of Chemical Engineers
ALARA	As Low as Reasonably Achievable
ALI	Annual Limit on Intake
AMSQ	Office of Assistant Manager for Environmental, Safety, Health, and Quality
ANS	American Nuclear Society
ANSI	American National Standards Institute
API	American Petroleum Institute
ARCHIE	Automated Resource for Chemical Hazard Incident Evaluation
ARF	Airborne Release Fraction
ASC	ALARA Subcommittee
ASME	American Society of Mechanical Engineers
BARCT	Best Available Radionuclide Control Technology
BAT	Best Achievable Technology
BDC	Baseline Design Criteria
BEI	BNFL Engineering Inc., United Kingdom
BIO	Basis for Interim Operation
BNFL Inc.	British Nuclear Fuels Limited Inc.
CAA	Clean Air Act
CAN	Corrective Action Notice
CAP	Corrective Action Plan
CAR	Construction Authorization Request
CBA	Cost Benefit Analysis
CCB	Consumable Changeout Box
CCPS	Center for Chemical Process Safety
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CLW	Co-Located Worker
CNWRA	Center for Nuclear Waste Regulatory Analysis
COCO	Company-Owned, Company-Operated
DAC	Derived Air Concentrations
DBE	Design Basis Earthquake
DCS	Distributed Control System
D&D	Decontamination and Decommissioning
DEAR	Department of Energy Acquisition Regulations
DF	Decontamination Factor
DID	Defense-in-Depth
DNFSB	Defense Nuclear Facilities Safety Board
DOE	United States Department of Energy
DSF	Design Safety Features
DST	Double Shell Tanks
DWPF	Defense Waste Processing Facility
EDG	Emergency Diesel Generator
EH	Office of Environment, Safety, and Health/DOE
EIS	Environmental Impact Statement
EM	Office of Environmental Management/DOE
EP	Environmental Protection OR Emergency Plan
EPA	U.S. Environmental Protection Agency
ERDF	Environmental Restoration Disposal Facility
ERPG	Emergency Response Planning Guideline
ERPP	Environmental Radiation Protection Program
ES&H	Environment, Safety, and Health
FFP	Firm Fixed Price (Contract)
FFRDC	Federally Funded Research and Development Center
FHA	Fire Hazards Analysis
FTE	Full Time Equivalent
GAO	General Accounting Office
GDP	Gaseous Diffusion Plant
GIGO	Garbage-In, Garbage-Out
GOCO	Government-Owned, Contractor-Operated
GTCC	Greater than Class C
HAB	Hanford Advisory Board
HAR	Hazards Analysis Report
HAZOP	Hazards Operability (method or analysis)
HEPA	High Efficiency Particulate Air
HFD	Hanford Fire Department
HLW	High Level Waste
HVAC	Heating, Ventilation, and Air Conditioning
HWMA	Hazardous Waste Management Act
I&C	Instrumentation and Control
IDLH	Immediate Danger to Life and Health
IEEE	Institute of Electrical and Electronics Engineers
IG	Implementation Guide
INEEL	Idaho National Engineering and Environmental Laboratory
IROFS	Items Relied on for Safety
ISA	Integrated Safety Analysis
ISAR	Initial Safety Analysis Report
ISM	Integrated Safety Management
ISMP	Integrated Safety Management Plan
ISO	International Organization for Standardization
ITS	Important to Safety
LAW	Low Activity Waste
LCAR	Limited Construction Authorization Request
LCO	Limiting Condition of Operation
LLW	Low Level Waste
LMAES	Lockheed Martin Advanced Environmental Systems
LNT	Linear No Threshold

MC&A	Material Control and Accounting
MCCMT	Miscellaneous Cold Chemicals Mix Tank
MI	Mechanical Integrity
M&O	Management and Operations (contract)
MOA	Memorandum of Agreement
MOU	Memorandum of Understanding
MOX	Mixed Oxide
NDT	Nondestructive Testing
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NESHAPS	National Emissions Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NIOSH	National Institute for Occupational Safety and Health
NMSS	Office of Nuclear Material Safety and Safeguards/NRC
NRC	U. S. Nuclear Regulatory Commission
NRPB	National Radiation Protection Board
OAR	Operations Authorization Request
ORP	Office of River Protection/DOE
OSHA	Occupational Safety and Health Administration
OSR	Office of Safety Regulation/DOE OR Operational Safety Requirements
PEL	Permissable Exposure Levels
P&ID	Piping and Instrumentation Drawings
PPE	Personnel Protective Equipment
PRA	Probabilistic Risk Analysis OR Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSHA	Preliminary Seismic Hazard Analysis
QA	Quality Assurance
QAP	Quality Assurance Program
QAPIP	Quality Assurance Program Implementation Plan
QC	Quality Control
RCM	Radiological Control Manual
RCRA	Resource Conservation Recovery Act of 1976
RF	Respirable Fraction
RFP	Request for Proposals
RG	Regulatory Guide
RIPB	Risk-Informed Performance Based
RL	Richland Operations Office/DOE
RMP	Risk Management Program
RO	Regulatory Official
RPP	River Protection Project/DOE OR Radiation Protection Plan
RPP/WTP	River Protection Project/ Waste Treatment Plant
RU	Regulatory Unit/DOE
SAP	Standards Application Package OR Standards Approval Package
SCR	Selective Catalytic Reduction
SER	Senior Expert Reviewers OR Safety Evaluation Report
SGI	Safeguards Information
SIP	State Implementation Plan
SIS	Safety Instrumentation Systems
SNF	Spent Nuclear Fuel
SNM	Special Nuclear Material
SQEP	Suitably Qualified and Experienced Person
SRD	Safety Requirements Document
SRP	Standard Review Plan
SRS	Savannah River Site
SSC	Structures, Systems, and Components
SST	Single Shell Tanks
TAN	Test Area North
TBD	To Be Determined
TEDE	Total Effective Dose Equivalent
TLV	Threshold Limit Value
TPA	Tri-Party Agreement
TRU	Transuranic (waste or isotopes)
TSD	Treatment, Storage, and Disposal
TWA	Time Weighted Average
TWRS	Tank Waste Remediation System
TWRS-P	Tank Waste Remediation System-Privatization
TWRS/ WTP	Tank Waste Remediation System/ Waste Treatment Plant
UBC	Uniform Building Code
UCNI	Unclassified Controlled Nuclear Information
UF	Ultrafiltration
WAC	Washington Administrative Code
WDOH	State of Washington Department of Health
WPPSS	Washington Public Power Supply System
WTP	Waste Treatment Plant
WVDP	West Valley Demonstration Project

GLOSSARY OF TERMS

BNFL Inc.	British Nuclear Fuels Limited, Inc., the U.S. subsidiary of BNFL plc and a TWRS-P contractor for Phases IA and IB-1. Often referred to as the "Contractor" in this report.
"Contractor"	This term refers to a specific, TWRS-P contractor (usually BNFL Inc.) in this report.
Defensive Waste Processing Facility	A high level waste vitrification plant at the Savannah River Site
Office of Environmental Management	DOE Headquarters Office responsible for DOE Complex-wide environmental and waste management activities.
High Level Waste	A term used to describe special nuclear fuel, first-cycle special nuclear fuel processing wastes and concentrates, and (for Tank Waste Remediation System) the solid phases in the tank wastes. High level waste requires disposal in a geologic repository.
Incidental Waste	A radioactive waste stream(s) comprised primarily of contaminated materials produced incidental to high level waste processing, such as spent resins, loaded filters, broken melters and equipment, and treated low activity waste. If the radiation levels are sufficiently low (generally interpreted as meeting the criteria for low level waste in 10 CFR Part 61), incidental waste may be treated and sent to disposal in near-surface facilities.
Low Activity Waste	A term used by DOE to describe the predominantly liquid portion of tank waste. Untreated low activity waste (LAW) is similar to high level waste in terms of environment, safety and health effects and requires geologic disposal. Treated LAW may be capable of meeting incidental waste criteria and, thus, it may be suitable for near surface disposal like low level waste.
Low Level Waste	A term for radioactive wastes that pose significantly lower radiological risks and of relatively short duration, such that the wastes are generally suitable for near-surface disposal per 10 CFR Part 61.
Richland Operations	A DOE office, located adjacent to the Hanford site. This office reports to the DOE Office of Environmental Management.
River Protection Project	DOE, Office of River Protection's program for activities that protect the Columbia River.
Regulatory Unit	Office of Safety Regulation of the River Protection Project/Waste Treatment Plant Contractor—DOE element responsible for regulating

the treatment and vitrification facilities. This office reports to DOE's Richland Operations Office.

Tank Waste Remediation System-Privatization	Refers to the private capital financed processing and vitrification facilities under the previous contracts held by BNFL Inc. and Lockheed Martin Advanced Environmental Systems.
Tank Waste Remedia tion System-Privatization /Waste Treatment Plant	Referring to the processing and vitrification facilities that will be built under the new contract(s) at Hanford.

1.0 INTRODUCTION AND BACKGROUND

1.1 HANFORD AND THE TANK WASTES

The U.S. Department of Energy (DOE) established the Tank Waste Remediation System (TWRS) program at the Hanford site to manage, retrieve, treat, encapsulate/immobilize, and disposition radioactive waste materials from the 177 underground waste storage tanks onsite in a safe, environmentally sound, and cost effective manner. These tanks primarily contain high level wastes (HLW) and chemical species from processing spent nuclear fuels for more than 40 years at the site (see Chapter 7.0, Main Reference 1). There are 149 single shell tanks (SSTs) and 28 double shell tanks (DSTs). There are several tank sizes but the average tank has about one million gallons of capacity. Both SSTs and DSTs are manufactured from carbon steels. However, the DSTs are newer, have more provisions for monitoring the wastes, and include an annulus for leak detection and confinement. To date, no DST has been confirmed to leak. In contrast, approximately 67 SSTs have been confirmed as leakers.

The tank contents consist of mixtures of materials from some eight major processes. Some of the wastes date back to 1944. Even though the radiation levels are high (typically exceeding 100 R/hr in the tank dome spaces and through riser connections), the great majority of the waste constituents are nonradioactive and contain some 240,000 tonnes of processed chemicals. The tanks hold approximately 54 million gallons of waste and amount to over 200 million-plus curies of radioactivity, primarily from cesium and strontium but with smaller contributions from other fission products and transuranic (TRU) isotopes. Physically, the tank contents exist as liquids, sludges, salts, saltcakes, and mixtures thereof, and some tanks periodically release gas mixtures. The SSTs contain primarily sludges and saltcakes with relatively little liquids - most of the liquid phase has been removed due to concerns about potential leaks. The DSTs contain most of the liquids but also have solid phases. The wastes stored in the tanks are defined as high level waste (HLW; per 10 CFR Part 50, Appendix F) and hazardous waste (per RCRA - Resource Conservation and Recovery Act - with various codes).

DOE categorizes the wastes to simplify contractual and processing approaches[1]. DOE uses the term LAW to denote "Low Activity Waste." Table 1 presents summary information on the composition of LAW. LAW is predominantly a liquid phase with soluble species such as nitrates and cesium; it may also contain up to 2 percent suspended solids or solids otherwise entrained by the waste transfers. Three envelopes of LAW have been defined; Envelope A is "standard," Envelope B contains higher levels of cesium, and Envelope C contains higher levels of strontium and TRU. The contract (Footnote 1) identifies ranges for chemical and radioactive species in these LAW envelopes. LAW would come from the liquid phases of the DSTs and from solids washing operations. From a regulatory perspective, LAW is still HLW and has high radiation levels requiring handling within shielded structures. DOE identifies the solid phases as "HLW," defined as Envelope D. Table 2 provides summary compositional information on HLW. Envelope D contains cesium, strontium, and TRUs as the radionuclides. Metal oxides, hydroxides, nitrates, phosphates, and aluminates constitute the bulk of the chemical species. The contract (Footnote 1) provides ranges for the chemical and radioactive species in Envelope

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[1] See, for example, Department of Energy (U.S.) (DOE). Contract No. DE-RP-96RL13308, "TWRS Privatization." DOE: Richland, Washington. August 1998.

D. Envelope D is assumed to be transferred as a slurry in concentrations up to 20 percent, from the removal of solid phases from the SSTs and DSTs. The solids in the LAW envelopes would have a composition similar to Envelope D.

Figure 1 provides a conceptual overview of DOE's approach to tank waste treatment. LAW envelopes would be transferred to a treatment plant. The LAW would be pretreated to separate the radionuclides (primarily cesium, strontium, technetium, and TRU, and the suspended solids) from the remainder of the waste envelope. The separated radionuclides would be stored for an interim period of several years. Pretreatment reduces the level of radioactivity in the treated LAW to levels commensurate with near-surface disposal requirements (essentially equivalent to the Class A/B/C definitions of low level waste in 10 CFR Part 61). The less radioactive, treated LAW would be vitrified and placed into stainless steel containers for long term storage or disposal at Hanford. The HLW (Envelope D) would be treated and washed using a filter or other device to separate the liquid phase from the slurry. The liquid phase would be routed to pretreatment and combined with the LAW, primarily for cesium and technetium removal. The treated HLW would be combined with the separated radionuclides from LAW processing and vitrified in an HLW melter. The HLW glass would be stored at Hanford in stainless steel canisters until subsequent disposal in an HLW repository.

Table 1: Summary Information on LAW Radionuclide Composition

Radionuclide	Maximum Ratio, Bq/mole Sodium			Curies/Liter at 10 Molar Sodium		
	Envelope A	Envelope B	Envelope C	Envelope A	Envelope B	Envelope C
TRU	4.8E5	4.8E5	3.0E6	1.3E-04	1.3E-04	8.11E-04
Co-60	6.1E4	6.1E4	3.7E5	1.65E-05	1.65E-05	1.0E-04
Sr-90	4.4E7	4.4E7	8.0E8	1.19E-02	1.19E-02	2.16E-01
Tc-99	7.1E6	7.1E6	7.1E6	1.92E-03	1.92E-03	1.92E-03
Cs-137	4.3E9	2.0E10	4.3E9	1.16E+00	6.00E+00 (contract max.)	1.16E+00
Eu-154 + Eu-155	1.2E6	1.2E6	4.3E6	3.24E-04	3.24E-04	1.16E-03

No contribution from the suspended and entrained solids is included in this table. LAW envelopes may contain up to 2 percent solids, which are assumed to be HLW solids. The solids contribution to radiotoxicity is significant and amounts to approximately 90 percent of the unit liter dose.

Table 2: Summary Information on HLW Radionuclide Composition (Reference 6)

Isotope	Ci/liter	Isotope	Ci/liter	Isotope	Ci/liter
H-3	1.30E-04	Cd-115m	(NS)	Eu-152	9.60E-04
C-14	1.30E-05	Sn-119m	(NS)	Eu-154	1.04E-01
Fe-55	(NS)	Sn-121m	(NS)	Eu-155	5.80E-02
Ni-59	(NS)	Sn-126	3.00E-04	U-233	1.80E-06
Co-60	2.00E-02	Sb-124	(NS)	U-235	5.00E-07
Ni-63	(NS)	Sb-126	(NS)	U-236	(NS)
Se-79	(NS)	Sb-126m	(NS)	U-238	(NS)
Sr-90	2.00E+01	Sb-125	6.40E-02	Np-237	1.48E-04
Y-90	(NS)	Te-125m	(NS)	Pu-238	7.00E-04
Nb-93m	(NS)	I-129	5.80E-07	Pu-239	6.20E-03
Zr-93	(NS)	Cs-134	(NS)	Pu-240	(NS)
Tc-99	3.00E-02	Cs-135	(NS)	Pu-241	4.40E-02
	(NS)	Cs-137	2.00E+01	Pu-242	(NS)
Rh-106	(NS)	Ba-137m	(NS)	Am-241	1.80E-01
Pd-107	(NS)	Ce-144	(NS)	Am-242	(NS)
Ag-110m	(NS)	Pr-144	(NS)	Am-242m	(NS)
Cd-113m	(NS)	Pr-144m	(NS)	Am-243	(NS)
In-113m	(NS)	Pm-147	(NS)	Cm-242	(NS)
Sn-113	(NS)	Sm-151	(NS)	Cm-243/244	6.00E-0

(NS) = Not Specified in the new contract.

Feed concentration contains between 10 and 200 grams of unwashed solids per liter of solution. Values in the table are based upon the upper limit of 200 grams/liter, which is approximately a 20% slurry (the actual value is closer to 15 percent).

1.2 THE TRI-PARTY AGREEMENT

The Hanford Federal Facility Agreement and Consent Order (also known as the Tri-Party Agreement, or TPA)[2] is a legal agreement between the DOE, the U.S. Environmental Protection Agency (EPA), and the State of Washington Department of Ecology (usually referred to as

[2] Environmental Protection Agency (U.S.)(EPA). EPA Docket Number 1089-03-04-120; Ecology Docket Number 89-54; "Hanford Federal Facility Agreement and Consent Order." EPA: Washington, D.C. May 15, 1989.

Ecology). The legal authority for the TPA arises from Resource Conservation Recovery Act of 1976 (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). The TPA contains provisions for the overall environmental management of the Hanford site, and includes provisions for the management of hazardous waste treatment, storage, and disposal (TSD) units and related permitting requirements. Consequently, the TPA incorporates provisions for TSD of the tank wastes. The TPA defines the respective roles, responsibilities, and interrelationships between the three parties. It delineates authorities, identifies enforcement provisions, and provides for dispute resolution among the parties.

The TPA has an action plan for compliance that establishes milestones for the Hanford Site Cleanup. These milestones constitute a minimally acceptable level of progress. Failure to meet the milestones can result in lawsuits and fines against DOE. Currently, the most relevant milestones applicable to the tank wastes are:

1. M-50-04: Start hot operations of HLW pretreatment facility by June 30, 2008.

2. M-50-04-T01: Submit conceptual design of HLW pretreatment facility by March 31, 1998.

3. M-50-04-T02: Initiate definitive design of pretreatment facility by November 30, 1998.

4. M-50-04-T03: Start construction of HLW pretreatment facility by June 30, 2001.

5. M-51-00: Complete vitrification of Hanford HLW (tank waste) by December 31, 2028.

6. M-51-03: Initiate hot operations of the HLW vitrification facility by December 31, 2009.

7. M-51-03-T01: Submit conceptual design of the HLW vitrification facility by December 31, 1998.

8. M-51-03-T02: Initiate definitive design of the HLW vitrification facility by December 31, 1998.

9. M-51-03-T04: Complete construction of the HLW vitrification facility by December 31, 2007.

10. Milestone M-61-00: Complete pretreatment and immobilization of the Hanford LAW by December 2028.

These milestones make for a tight schedule. The current design would treat half the waste by circa 2050, while the DOE program desires to have all of the waste treated by 2028, a difference in processing capacity of at least a factor of four[3].

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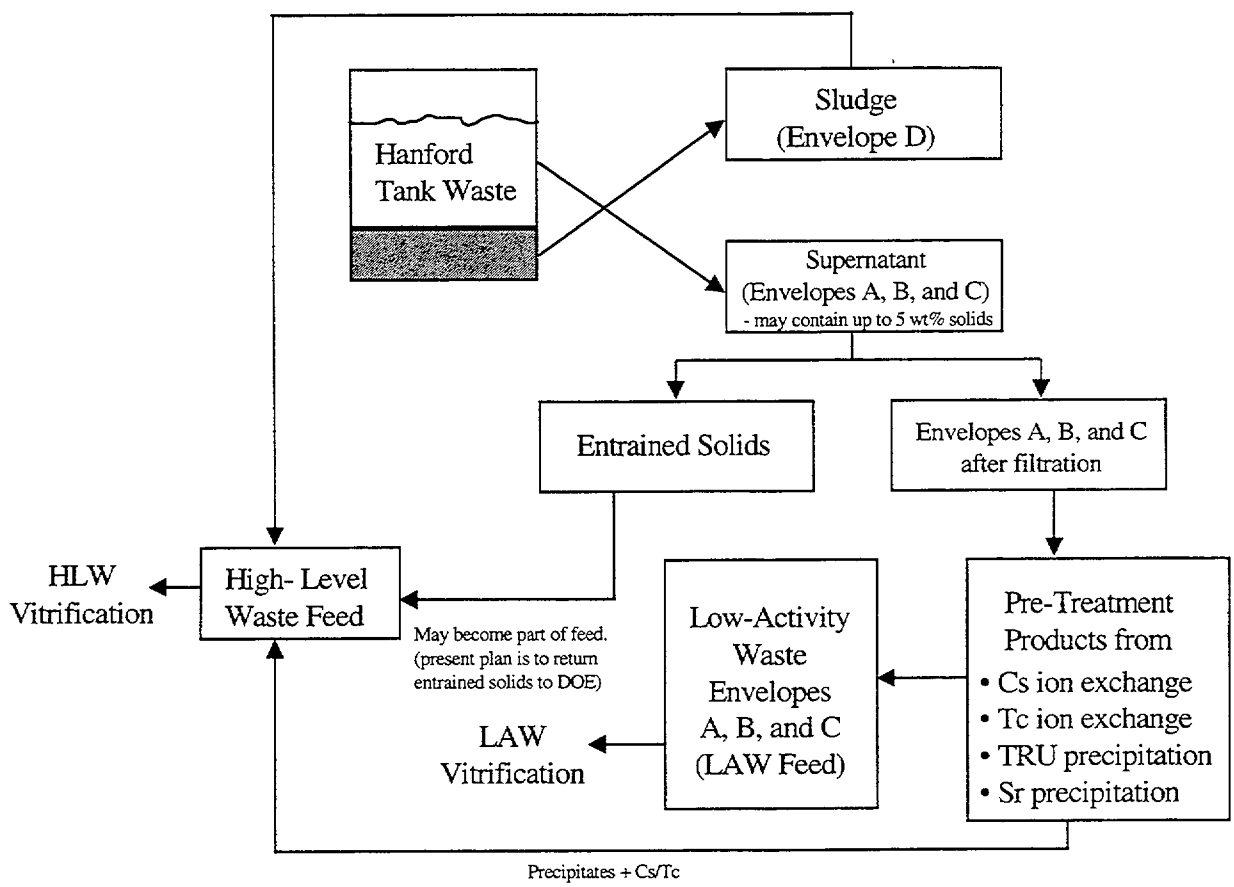
[3] "Waste Management 2000," presentations by Mike Lawrence, BNFL Inc., and Dick French, U.S. Department of Energy. March 2000.

Figure 1: Overview of Hanford Processing Approach



1.3 DOE PROGRAM AND CONTRACTS

DOE was pursuing a privatization initiative at Hanford for the construction and operation of contractor-owned, contractor-operated facility or facilities for treating these tank wastes and meeting the TPA milestones[4]. The concept was for DOE to enter into two or more firm fixed-price (FFP) contracts for the contractor(s) to build and operate a facility to treat the tank wastes according to DOE requirements. A minimum of two contractors would ensure that the government would receive the lowest price for waste TSD. The TWRS-Privatization Program was divided into two phases. Phase I was a proof-of-concept, commercial demonstration scale effort with the following objectives:

1. Demonstrate the technical and business viability of using privatized contractors to treat Hanford tank wastes.

2. Define and maintain adequate levels of radiological, nuclear, and process safety.

3. Maintain environmental protection and compliance.

4. Substantially reduce life-cycle costs and time for treating the wastes.

The original plan called for processing between 6 and 13 percent of the tank waste in a pilot/demonstration facility as Phase I, and a subsequent, larger program would process the balance of the tank wastes as Phase II. Phase I consisted of two parts. Part A consisted of a 20-month development period to establish appropriate and necessary technical, operational, regulatory, business, and financial elements. This required the privatization contractors to select safety standards and requirements, formulate integrated safety management plans, and to generate conceptual designs and initial safety analyses, all of which would require approval by DOE. The contractors worked for 16 months to develop these items and the remaining 4 months were used by DOE for evaluation.

DOE awarded contracts of $27 Million (each) for Part A to two teams, one led by British Nuclear Fuels Limited Inc. (BNFL Inc.), and the other led by Lockheed Martin Advanced Environmental Systems (LMAES). The BNFL Inc. team proposed a conceptual approach based upon the following operations (Chapter 7.0, Main References 2-5):

1. Strontium/TRU coprecipitation from LAW.
2. Suspended solids/strontium/TRU removal by ultrafiltration from LAW.
3. Two columns in series, organic ion exchange recovery of cesium from LAW.
4. Two columns in series, organic ion exchange recovery of technetium from LAW.
5. Optional loading of radiocesium onto crystalline silicotitanate (CST).
6. LAW vitrification in a joule-heated melter.
7. HLW washing and concentration by ultrafiltration.
8. HLW vitrification in a joule-heated melter.
9. NOx treatment by selective catalytic reduction (SCR) using anhydrous ammonia.

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[4] Department of Energy (U.S.) (DOE). RL/RU-2000-20, Rev. 0, "Regulatory Unit Position on Important to Safety Work Authorization for the RPP-WTP Interim Design Period." DOE: Richland, Washington. July 3, 2000.

Some aspects of the conceptual design were more detailed than other parts. For example, each ion exchange column would enclose 3.15 cubic meters, with an aspect ratio of about 7. In use, the columns would only contain about one cubic meter of resin each, for an effective aspect ratio of about 1.5-2. The approach included four ion exchange columns for cesium removal, arranged as two trains of two columns each. The technetium arrangement was identical. The design included two HLW and three LAW melters, and around 300,000 gallons of tankage.

In terms of technology, BNFL Inc. relied upon proprietary organic ion exchange resins ("superligands") and ultrafilters for separation of the radionuclides and solids from the LAW in pretreatment. Ultrafiltration was also planned as part of the treatment of HLW. BNFL Inc. proposed a liquid-fed slurry ceramic melter with joule heating for HLW vitrification, similar to the approach used at the West Valley Demonstration Project (WVDP) in New York State for vitrifying that site's HLW. LAW vitrification also used a joule-heated melter for the proposed facility, but it would be considerably larger than the HLW design. As an alternative, BNFL Inc. proposed storage and return of suspended solids to DOE without HLW vitrification; the cesium would be returned on CST. BNFL Inc. identified technology development in order to address unknowns about the new technologies involved.

LMAES proposed a similar conceptual process approach with several variations (see Chapter 7.0, Main References 6-9):

1. Suspended solids removal from LAW by centrifugation.

2. Three columns in series, organic ion exchange recovery of cesium from LAW, regenerated by nitric acid and caustic.

3. Three columns in series, organic ion exchange, polishing step recovery of cesium (from the effluent of the preceding step), with electrical regeneration.

4. An inorganic, "guard" bed for cesium removal (on the effluent of the preceding step).

5. Optional loading of radiocesium onto CST.

6. Removal of technetium from LAW by electroplating.

7. Strontium and TRU removal using ozone destruction of organics followed by precipitation and centrifugation.

8. LAW vitrification in a joule-heated melter, augmented by fired burners (based upon oxygen-propane combustion) during startup and glass pouring.

9. HLW washing and concentration by centrifugation.

10. HLW vitrification in a cold-crucible, induction-heated melter.

11. NOx treatment using SCR and ammonia from aqueous solution.

Each ion exchange column would be approximately 0.67 cubic meters, containing about 0.6 cubic meters of resin, with a working aspect ratio of approximately four. The approach effectively used six columns in series to remove the radioactive cesium. The design incorporated one HLW and three LAW melters, and around 200,000 gallons of tankage.

In terms of technology, ion exchange technology was planned for cesium separation from LAW, using resins tested by Savannah River Site (SRS), with CST used as a guard column. Centrifuges separated suspended solids from the LAW. Cold crucible induction melting, similar to the process at La Hague (France) was planned for HLW vitrification. For LAW vitrification, LMAES proposed to use a direct fired melter, undergoing development on other types of DOE wastes. The approach planned to use electrolysis for technetium removal (by plating) and for recycling some of the ion exchange regeneration solutions.

Phase I, Part A has been completed; the contractors each submitted a System Requirements Document (on standards), Hazards Analysis Report, Integrated Safety Management Plan, and Initial Safety Analysis Report. The DOE generated safety evaluation reports on the contractor submittals (Main References 10 and 11), indicating many concerns and open issues on each contractor's approach. For LMAES, the DOE review team concluded that the approach described in the ISAR would be capable of achieving subsequent authorizations for construction, operation, and deactivation, provided some 37 open issues would be resolved in the next regulatory submittal (Construction Authorization Request (CAR) — essentially a Preliminary Safety Analysis Report (PSAR)). Of these 37 issues, development of the pretreatment technology to maturity and its subsequent safe implementation were identified as the most significant uncertainties. For BNFL Inc., the DOE review team concluded that no individual regulatory issue would prevent subsequent safety authorizations. However, the review identified 90 representative open regulatory issues that would require resolution, and the nature and number greatly challenged the reviewers to reach a consensus on the viability and sufficiency of some of the approaches the Contractor proposed to achieve and maintain adequate safety through design and management practices.

In May 1998, DOE said BNFL Inc. had presented a superior plan for TWRS-P because the LMAES proposal posed an "unacceptably high technical risk" of failure to meet DOE's cleanup goals[5]. In September 1998, DOE entered into a revised contract with BNFL Inc. In addition to modifying the original intent of two contractors and competition, DOE changed the program to accommodate larger production scale facilities in Phase I that would have a 30-40 year useful life. Phase I, Part B-1 involved a 24-month facility design phase that would advance the design to approximately a "30 percent level" and have the Contractor prepared to start construction and obtain financing[6]. Furthermore, in this 24-month period, DOE and BNFL Inc. were to refine the technical requirements, submit regulatory permitting applications, and finalize fixed unit prices for treated wastes. This Phase I, Part B-1 was estimated to be worth approximately $350 million, using a fixed fee type of contract. Facility construction and operation was to occur in a subsequent, planned Phase I, Part B-2. For Part B-2, the Contractor would only receive payment for waste actually processed and vitrified. This part of the effort was projected to cost DOE approximately $6.9 billion (see Footnote 4, page 22). In total, Part B was expected to

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[5] See, for example, *The Energy Daily*, May 22, 1998 and *Inside Energy/with Federal Lands*, May 25, 1998.

[6] U.S Department of Energy/Hanford Press Release, July 21, 1998.

require 5-8 years for design and construction activities, and another 5-10 years for the processing of the initial, 6-13 percent of the tank waste.

At the time of this writing, Phase II planned to enlarge and utilize the Phase I facilities instead of constructing and using entirely new facilities. Hence, the proposed Phase I facilities would be relatively large. This diminished the distinction between the two phases and implied that the regulatory framework for Phase I would continue into Phase II. The Phase II activities extended the waste processing time frame by another 10-30 years, depending upon the facility capacities, waste retrieval activities, and difficulties encountered. Potentially, plant operation could continue until circa 2050.

DOE anticipated the total value of the Part B work to be in the $7-10 billion range[7] (also see Main Reference 12). As part of the major design submittal (Main Reference 13), BNFL Inc. identified potential costs as high as $15 billion if the privatization route was continued. On May 8, 2000, DOE decided to stop the privatization initiative for Hanford tank waste primarily for economic reasons, and elected to terminate the BNFL Inc. contract (Footnotes 5-7). Project management and quality assurance concerns were also mentioned as reasons for contract termination. The program name was also changed from Tank Waste Remediation System (TWRS) to River Protection Project (RPP). While the specific details are still evolving, DOE is using a transition contractor (CH2MHILL Hanford Group) to continue the design efforts while a competitive procurement is conducted for a new contractor for completion of the design and construction of the facility. A separate, additional contract would be released for facility operations. All of these new contracts would utilize M&O (management and operations) style contracts with cost-reimbursement and incentive clauses.

1.4 DOE REGULATORY APPROACH

DOE is a self-regulating agency on nuclear safety matters. DOE's goal in proceeding with the radiological, nuclear, and process safety regulation of TWRS-P contractors is to establish a regulatory environment that will permit privatization to occur on a timely, predictable, and stable basis with attention to safety consistent with that which would accrue from regulation by an external agency[8]. This same policy states that DOE is patterning its regulation of TWRS-P contractors to be consistent with the NRC's regulatory approach. The Manager of the Richland Operations Office (DOE/RL) has the responsibility for safety of activities on the Hanford site. The policy and the related Memorandum of Agreement (MOA)[9] established a dedicated Regulatory Unit (RU) led by a Regulatory Official (RO) at the DOE Richland Operations Office with regulatory authority exclusive to the regulation of TWRS-P contractors. The RO would report directly to the Manager of DOE/RL at a level equivalent to the DOE Program Manager for TWRS. In implementation, the RU is to follow the five principles of good regulation as

[7] See Footnote 4 of this document and *The Energy Daily*, May 15, 2000

[8] Department of Energy (U.S.)(DOE). DOE/RL-96-25, Rev. 0, "Policy for Radiological, Nuclear, and Process Safety Regulation of TWRS Privatization Contractors." DOE: Richland, Washington. July 3, 1996.

[9] Department of Energy (U.S.)(DOE). DOE/RL-96-26, Rev. 0, "Memorandum of Agreement for the Execution of Radiological, Nuclear, and Process Safety Regulation of TWRS Privatization Contractors." DOE: Richland, Washington. July 3, 1996.

articulated by the NRC - independence, openness, efficiency, clarity, and reliability. The RU has four organizational elements: the Standards and Requirement Group, the Activities Authorization Group, the Verification and Confirmation Group, and the Senior Technical Team. The Manager of DOE/RL also has a three-member, Senior Expert Review (SER) panel for periodic assessment of the RU and major issues.

The basic concept of DOE's regulatory approach is that the contractor is responsible for achieving adequate safety, complying with applicable laws and regulations, and conforming with top-level safety standards and principles stipulated by DOE[10]. Consistent with applicable laws and regulations, the contractor is required to tailor the exercise of this responsibility to the specific hazards associated with its activities, and is encouraged to do this in a cost-effective manner that applies best commercial practices. TWRS-P contractors have the responsibility to identify and recommend to DOE the set of standards, regulations, and requirements necessary to ensure adequate safety. DOE's responsibility is to execute the regulatory process, including authorization of contractor actions and confirmation that the contractor activities are performed safely and within approved limits.

The authority of the RU to regulate a TWRS-P contractor is derived from the terms of the TWRS-P contract ("regulate by the contract"). In addition to the regulatory concept, the following three radiological, nuclear, and process safety related documents are incorporated into the contract:

1. "DOE Process for Radiological, Nuclear, and Process Safety Regulation of the TWRS-P Contractor," DOE/RL-96-0003.

2. "Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for the TWRS-P Contractor," DOE/RL-0006.

3. "Process for Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for the TWRS-P Contractor," DOE/RL-96-0004.

The two, nonradiological documents are:

1. "Industrial Hygiene and Safety Regulatory Plan," RL/REG-2000-04.

2. "Regulatory Unit Position on Regulation of the Contractor's Industrial Hygiene and Safety Program," RL/REG-99-11.

The RU has further explained the regulatory process as "bottoms up," starting with the contractor establishing the set of standards needed to achieve safety, comply with applicable laws, achieve the DOE top-level standards and principles (DOE/RL-96-006), and follow an

[10] Department of Energy (U.S.)(DOE). DOE/RL-96-0005, Rev. 1, "Concept of the DOE Regulatory Process for Radiological, Nuclear, and Process Safety for TWRS Privatization Contractors." DOE: Richland, Washington. July, 1998.

integrated safety management process (ISMP, in DOE/RL-96-0004)[11]. The regulator would subsequently approve the contractor's set of standards. A clear, central concept of ISM is that the contractor should tailor the design and safety requirements to the specific hazards of the activities and operations at a facility. DOE/RU policy endorses tailoring via the following process:

1. Identify applicable requirements.
2. Define the scope of the work or operations to be analyzed.
3. Analyze the hazards.
4. Propose, analyze, select, and implement controls.
5. Perform the work or operations (does not apply at the design stage).
6. Assess, feedback, and improve/modify (as appropriate).

The Contractor conducted two ISMP cycles on the TWRS-P design prior to contract termination.

DOE defined the following, minimum set of regulatory process elements:

1. The top-level standards and principles.
2. Standards identification (including the ISMP).
3. DOE review and approval.
4. Initial safety analysis and review.
5. Construction authorization.
6. Operating authorization.
7. Regulatory oversight, including inspection of design, construction, or operating activities.
8. Deactivation authorization.
9. Independent oversight of TWRS regulation by DOE.
10. Public information and involvement.

(Note that the contract was terminated in the design phase prior to the CAR.)

The RU has published guidance, policy, and position documents to assist its staff in these regulatory elements. Of these, the CAR review guidance (Main Reference 14) would be used in a manner similar to that of a standard review plan. This guidance was used in the review of the Firm Fixed Price submittal (Chapter 7.0, Main Reference 13).

The RU also established an inspection program for the Phase IB-1 activities. As of this writing, thirteen inspections were conducted over the 20-month period.

DOE conducted two external assessments of the RU regulatory program and related activities. DOE, Office of Environment, Safety, and Health (EH), assessed the readiness of the RU in 1998[12]. The EH team reviewed TWRS-P related documents and interviewed personnel from

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[11] Department of Energy (U.S.)(DOE). RL/REG-98-21, "Regulatory Unit Position on Implementing and Assuring Compliance with Integrated Safety Management." DOE: Richland, Washington. August 26, 1998.

[12] Department of Energy (U.S.)(DOE). DOE/EH-0569, "Assessment of Tank Waste Remediation System-Privatization Regulatory Unit Readiness." DOE: Richland, Washington. April 1998.

the RU, DOE/RL, DOE-Headquarters, public, stakeholders, and the two Phase IA contractors. The review identified several weaknesses in the areas of management and organization, interfaces, staffing, technical standards and requirements, authorization process (contract consistency), document reviews, and inspection and enforcement. One weakness noted that the regulatory approach with two contractors resulted in two ISMP systems, different deliverable formats, and different submittal schedules. The RU subsequently developed a corrective action plan in response to these weaknesses[13]. The RU only disagreed with the finding of one weakness; review schedule pressures. The EH team expressed concerns about the lack of specific provisions for delaying a review schedule when it cannot be supported by the RU. The EH team noted that RU directives and DOE documentation confirm the RU's responsibilities for meeting schedules, except when the contractor information is insufficient. The EH team further stated that review schedule pressures may cause reviewers to be less thorough, may compromise safety, and may give the public the impression that production takes precedence over safety and that the RU is not fully independent. In response, the RU noted that the revised contract adequately protects both the DOE and the Contractor from delays in the schedule caused by the extension of reviews.

A second external assessment of the RU and the regulatory process was conducted by the Senior Expert Reviewers (SER)[14]. The SER panel reviewed documents, observed RU operations, and conducted interviews with four senior officials from DOE and BNFL Inc. (there were no interviews of NRC officials). The SER report considered the performance of the RU in planning and executing a first-of-a-kind regulatory concept exemplary. The report noted concerns about the level of detail, the Topical Meeting process, the need for submittal of any proposed changes in fundamental design safety features in the same time frame as the design finalization, and NRC reviews. On the latter, the SER panel concluded that the NRC reviews do not appear to have identified significant differences between the safety requirements invoked by the RU or the NRC, but noted a significant difference in the regulatory system being developed by the NRC staff for possible application to TWRS-P (essentially 10 CFR Part 70 and NUREG-1702), as compared to the DOE/RU regulatory system or process.

The Office of River Protection (ORP) was established at the Hanford site in December 1998. As directed by Congress, DOE is using ORP to focus management responsibility and accountability. ORP has assumed the former RL/TWRS role of overseeing and directing TWRS, TWRS-P, and activities related to the tank wastes. ORP reports directly to DOE Headquarters and not to DOE/RL. ORP manages the TWRS-P contract and regularly interacts with the contractors on program issues. At the time of this writing (December 2000), the RU has become part of the ORP and reports to ORP management. The RU has been renamed the Office of Safety Regulation (OSR), and the OSR is responsible for regulating the radiological, nuclear, and process safety of the facility being built to take waste currently stored in underground tanks at Hanford and process it to a glass form. From the new contract[15], the

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[13] Department of Energy (U.S.)(DOE). RL/REG-98-15, "Regulatory Unit (RU) Readiness Assessment (DOE/EH-0569) - RU Response and Corrective Actions." DOE: Richland, Washington. July 1998.

[14] Department of Energy (U.S.)(DOE). 00-RU-0005, "Report of an Assessment of the Regulatory Unit for the River Protection Project Privatization Contract," Senior Expert Reviewers, September 1999.

[15] Contract Number DE-AC27-01RV14136, "Bechtel National Inc.: Design, Construction, and Commissioning of the Hanford Tank Waste Treatment and Immobilization Plant."

contractor is required to follow an integrated safety management (ISM) process, with approvals from DOE (presumably the OSR). The ORP organization also includes an "Office of Assistant Manager for Environmental, Safety, Health, and Quality" (AMSQ). The actual missions and working relationships of OSR and AMSQ within the ORP are not clear at this time.

For the interim design period, the authority of the RU to regulate the RPP-WTP contractor was derived from a recent memorandum[16] and stated in an RU position paper[17].

1.5 NRC INVOLVEMENT VIA THE MEMORANDUM OF UNDERSTANDING

DOE also considered the potential for external regulation, and entered into a Memorandum of Understanding (MOU) with the Nuclear Regulatory Commission (NRC) for cooperation and mutual support, with the possibility of transitioning to NRC regulation sometime in the future Attachment B). The MOU had two main purposes:

1. The DOE to acquire the capability to implement a program of nuclear safety and safeguards regulation consistent with the NRC's regulatory approach.

2. The NRC to acquire sufficient knowledge and understanding of the physical and operational situation at the Hanford waste tanks and the processes, technology, and hazards involved in Phase I activities to enable the NRC (a) to assist DOE in performing reviews in a manner consistent with NRC's regulatory approach and (b) to be prepared to develop an effective and efficient regulatory program for the licensing of DOE contractor-owned and contractor-operated facilities that will process waste at Hanford during Phase II.

Prior to the termination of the privatization contract, DOE and the NRC were negotiating a revised MOU that recognized the reuse of the Phase I facilities for Phase II and maintained the NRC role in the near-term. However, the revised MOU implied a delay in regulatory transition, if it were to occur at all. In addition, the impact of the termination of the privatized contract and the use of new M&O-style contracts would have diminished or ultimately even eliminated NRC's participation, as it is likely that DOE will self-regulate the proposed facilities without transition to independent regulation by the NRC. Consequently, the NRC has decided not to participate in the near-term but has stated it is willing to discuss possible arrangements for NRC involvement in the TWRS project in the future, if there is again a need for NRC expertise and if the NRC sees benefit in such involvement, such as the potential transition to NRC regulatory authority.

1.6 OVERVIEW OF THIS DOCUMENT

Chapter 2.0 summarizes the NRC participation and activities.

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[16] Huntoon, C.L., U.S. Department of Energy, memorandum to R.T. French, U.S. Department of Energy, "Maintaining Safety and Quality through Contract Transitions." May 23, 2000.

[17] Department of Energy (U.S.)(DOE). RL/REG-2000-20, Rev. 0 "Regulatory Unit Position on Important to Safety Work Authorization for the RPP-WTP Interim Design Period." DOE: Richland, Washington. July 3, 2000.

Chapter 3.0 discusses areas of design and safety that the NRC would typically review from a potential license application on TWRS-P, and provides an assessment of the current status based upon the BNFL Inc. documentation through June, 2000.

Chapter 4.0 presents NRC observations and conclusions.

Chapter 5.0 summarizes potential issues for transition of TWRS-type facilities to NRC regulation.

The attachments provide information on the major technical issues from the perspective of the NRC staff, point paper summaries, the DOE/NRC MOU, potential issues for regulatory transition, and an index listing of correspondence between the NRC and DOE on the program.

2.0 NRC PARTICIPATION AND ACTIVITIES

2.1 TANK WASTE REMEDIATION SYSTEMS SECTION

In order to support the Memorandum of Understanding, the U.S. Nuclear Regulatory Commission (NRC), in October, 1996, established the Tank Waste Remediation System (TWRS) Section within the Special Projects Branch of the Fuel Cycle Safety and Safeguards Division, in the Office of Nuclear Materials Safety and Safeguards. This Section had the following specific tasks:

1. Provide technical support to the U.S. Department of Energy (DOE) for activities related to the DOE TWRS privatization (TWRS-P) activities at Hanford.

2. Develop sufficient knowledge and understanding of the physical and operational situation at the Hanford waste tanks and the processes, technologies, and hazards involved in Phase I activities to (a) assist DOE in performing reviews in a manner consistent with NRC's regulatory approach and (b) be prepared to develop an effective and efficient regulatory program for the licensing of DOE contractor-owned and contractor-operated facilities that will process Hanford wastes in Phase II.

3. Conduct safety, safeguards, and environmental reviews concerning TWRS-P operations and processes.

4. Review and comment on DOE regulations and guidance for the regulation of activities related to Phase I of the tank waste remediation activities at Hanford.

5. At the request of DOE, participate as appropriate in the development of guidance for use by DOE, including guidance based upon industry standards (e.g., American Nuclear Society, American National Standards Institute, American Society of Mechanical Engineers).

6. Review existing NRC regulations and guidance for their potential applicability to TWRS-P and its potential licensing in the future by NRC, including identification of potential modifications and additions.

7. Identify potential differences between the DOE and NRC regulatory approaches as they apply to TWRS-P and notify DOE of these potential differences.

To accomplish these tasks, the NRC recruited senior technical talent from both within and outside of the agency for the TWRS Section with experience in high level waste (HLW), HLW chemistry and processing, vitrification, Hanford, DOE, and regulatory activities. The following positions were filled:

1. Section Leader
2. Senior Onsite Technical Representative
3. Senior Chemical Process Engineer
4. Fire Protection Specialist

5. Senior Nuclear Process Engineer
6. Quality Assurance Specialist
7. Structural/Construction Engineer
8. Chemical Engineer
9. Metallurgical Specialist/Engineer
10. Nuclear Process Engineer
11. Chemical/Materials Engineer
12. Mechanical Engineer
13. Environmental Engineer
14. Health Physicist
15. Licensing Assistant

The staff had an average experience of approximately 20 years. Staff would rotate into Hanford for extended time periods in order to support the DOE-Regulatory Unit (RU), equivalent to a full time position onsite. In addition, a Senior Onsite Representative was located at Hanford, in the same office area as the RU. The onsite representative interacted with the RU on a daily basis, participating in meetings and presentations involving DOE and the contractors. The remaining staff were located in the Rockville, Maryland, offices of the NRC Headquarters. This staff visited the Hanford site many times for attending meetings and presentations by DOE and the contractors. An annual average of 57 trips were made in support of the RU and the TWRS-P program. Activities included review, analysis, and comment on contractor submittals, guidance documents, procedures, RU documents, etc. Usually, several members of the NRC Headquarters staff would rotate to Hanford for TWRS-related meetings for a week each month. Several major review activities required members of the staff to remain in the Richland area for extended time periods. The staff at NRC Headquarters also communicated frequently with their DOE counterparts via phone, E-mail, and teleconferencing. NRC staff accompanied RU personnel on inspections. NRC staff reviewed NRC regulations and guidance related to TWRS-P. A standard review plan was prepared, issued, and finalized (see Section 2.3). NRC staff attended conferences and training sessions related to TWRS technical areas, such as quality assurance and hazards analysis. Several NRC staff members also visited existing HLW processing facilities in Savannah River (Defensive Waste Processing Facility), West Valley (West Valley Demonstration Project), United Kingdom (Sellafield), and France (La Hague).

2.2 SUPPORT ACTIVITIES BY THE CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES

The Center for Nuclear Waste Regulatory Analyses (CNWRA) was established in 1987 as an NRC-sponsored, federally funded research and development center (FFRDC), in order to provide technical assistance and research in support of the NRC HLW program. Because of the similarities in the waste forms and processing requirements, the NRC decided to fund support work related to TWRS at the CNWRA. The CNWRA program had the following three main objectives:

1. Provide technical assistance to the NRC for developing a regulatory framework and associated guidance that would be used to license a TWRS-P facility.
2. Assist the NRC in the review of contractor submittals.
3. Develop an experience base for future activities in the program (e.g., construction and operations).

In support of these main objectives, the CNWRA effort had the following tasks:

1. Familiarization, regulatory development, and safety reviews.
2. Preoperational reviews of pilot processing facilities.
3. Operational reviews of pilot facilities and reviews of safety bases.
4. Revision of the regulatory framework.

The CNWRA primarily operated under Task 1 in support of the NRC. The CNWRA produced analyses and reports on Hanford chemistry, risk/safety analyses, waste solidification, and process experience. This included participation in site visits to operating vitrification facilities at Savannah River, La Hague (France), and Sellafield (United Kingdom). The CNWRA analyzed consequence criteria, potential doses and risks, and issues, such as fires, explosion, and radiolysis. The CNWRA also assessed TWRS-P chemistry and processes, including several reports on the tank wastes, vitrification, separations, and a spreadsheet model of pretreatment processes.

2.3 TWRS-P STANDARD REVIEW PLAN

As part of its activities, the NRC has published the NUREG-1702, "Standard Review Plan for the Review of a License Application for the Tank Waste Remediation System Privatization (TWRS-P) Project" (Chapter 7.0, Main Reference 15). This provides NRC guidance for the review and evaluation of health, safety, and environmental protection in applications for licenses for remediation of radioactive tank waste at Hanford. The guidance is also applicable to the review and evaluation of proposed amendments and license renewal applications. Specific filing requirements for license applications and for issuance of such licenses are in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," as revised[18]. Although 10 CFR Part 70, as revised, does not specifically include a TWRS-P facility in 10 CFR 70.60, "Applicability," the regulation specifies applicable facilities which include, "any other activity that the Commission determines could significantly affect public health and safety."

The principal purpose of the Standard Review Plan (SRP) is to ensure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope, level of detail, and acceptance criteria of reviews. The SRP also should be used as the basis for the review of requests by licensees for changes in their licenses. Thus, the SRP, at any point in time, can provide a basis for the review of proposed new or renewal applications, and amendments to existing licenses, as well as modifications to the SRP resulting from new NRC requirements and licensee initiatives.

Another important purpose of the SRP is to make information about regulatory reviews widely available and to improve communication and understanding of the staff review process. Because the SRP describes the scope, level of detail, and acceptance criteria for reviewers, it can serve as regulatory guidance for applicants who need to determine what information should be presented in a license application.

[18] Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

The responsibility of the staff in the review of a license application, renewal application, or license amendment for a TWRS-P facility is to determine that there is reasonable assurance that the facility can and will be operated in a manner that will not be inimical to the common defense and security, and will provide reasonable protection of the health and safety of workers and the public, and the environment. To carry out this responsibility, the staff evaluates information provided by an applicant and through independent assessments determines that the applicant has demonstrated a reasonable safety program that is in accordance with regulatory requirements. To facilitate carrying out this responsibility, the SRP clearly states and identifies those standards, criteria, and bases that the staff should use in reaching licensing decisions.

The staff believes that a TWRS-P facility is an activity that could significantly affect public health and safety. This belief is due to the presence of multi-kilogram quantities of special nuclear material (SNM) (slightly enriched uranium and plutonium), kilocurie quantities of transuranics (TRUs), megacurie quantities of fission products, and numerous chemical species in the wastes, and the need for a large plant with relatively high flow rates, large inventories, numerous process steps, and energetic operations. Application of the specific technologies and equipment in a highly radioactive environment would be new and at a magnitude beyond any other applications and without significant pilot plant testing, thus increasing the unknowns and uncertainties. The melters would become the largest of their respective types in the world and have unique features. Therefore, if the NRC became the regulator, it would plan to invoke the requirements found in Subpart H of 10 CFR Part 70 for this type of facility. As such, 10 CFR Part 70, as revised, requires that an applicant submit a complete description of the safety program for the possession and use of SNM to show how compliance with the applicable requirements will be accomplished. The Safety Program Description must be sufficiently detailed to permit the staff to obtain reasonable assurance that the facility is designed and will be operated without undue risk to the health and safety of workers or the public. Prior to submission of the program description, an applicant should have analyzed the facility in sufficient detail to conclude that it is designed and can be operated safely. The Safety Program Description is the principal document with which the applicant provides the information needed by staff to understand the basis for conclusion. When reviewed and approved by the staff, and incorporated in the NRC license by reference, the Safety Program Description, in its entirety and in its parts, is the safety basis on which the license is issued and may not be changed except through the process defined in 10 CFR 70.72.

The requirements in 10 CFR Part 70 specify, in general terms, the information to be supplied in a Safety Program Description. The specific information to be submitted by an applicant and evaluated by staff is identified in this SRP. Prospective applicants should study the topic areas treated in this document (generally, chapter headings) and the sections within each topic area, specifically the sections headed "Areas of Review" and "Acceptance Criteria." A license application should contain a Safety Program Description that addresses all the topics in the Table of Contents of this SRP, in the same order as presented in this document.

In this SRP, information is provided to assist the licensing staff and the applicant in understanding the underlying objective of the regulatory requirements, the relationships among NRC requirements, the licensing process, the major guidance documents NRC staff has prepared for licensing facilities under 10 CFR Part 70, and the details of the staff review process set out in individual SRP sections. Analyses by the staff are intended to provide regulatory confirmation of reasonable assurance of safe design and operation. A staff determination of reasonable assurance leads to a decision to issue or renew a license or to approve an amendment. In the case of a staff determination of inadequate description or commitments, the staff should inform the applicant of what is needed and the basis upon which the determination was made.

The "Acceptance Criteria" delineated in this SRP are intended to communicate the underlying objectives but not to represent the only means of satisfying that objective. An applicant should tailor its safety program to the features of its particular facility. If approaches different from the SRP are chosen, the applicant should identify the portions of its application that differ from the design approaches and acceptance criteria of the SRP and evaluate how the proposed alternatives provide an acceptable method of complying with the Commission's regulations. The staff retains the responsibility to make an independent determination of the adequacy of what is proposed.

The major topics addressed within the Safety Program Description of a facility license application are addressed in separate SRP chapters, as follows:

- Chapter 1: General Information
- Chapter 2: Organization and Administration
- Chapter 3: Integrated Safety Analysis
- Chapter 4: Radiation Safety
- Chapter 5: Nuclear Criticality Safety
- Chapter 6: Chemical Safety
- Chapter 7: Fire Protection
- Chapter 8: Emergency Management
- Chapter 9: Environmental Protection
- Chapter 10: Decommissioning
- Chapter 11: Management Measures
- Chapter 12: Plant Systems

The applicant's integrated safety analysis (ISA) is the central focus for the selection of design and operational safety measures and the management control systems that assure the availability and reliability of those measures. It is the ISA that provides a comprehensive evaluation and presentation, useful to both the applicant and the NRC, of the distribution of risk among the many activities ongoing at the TWRS-P facility. The NRC expects to be able to use the ISA summary to focus its resources on the dominant risks of facility design and operation and the safety controls and assurances necessary to ensure that those controls remain available and reliable. Accordingly, staff reviewers should conduct a coordinated review of the ISA summary and focus on the portions of the summary that are applicable to each of the technical areas treated in the chapters of the SRP. The acceptance criteria in each of the SRP chapters are the criteria that apply to the dominant risks of operation. The applicant has the opportunity to justify lesser criteria for those design and operational features that can be shown to represent lesser risk than the accident or failure sequences that pose the dominant risks.

While recognizing the fundamental importance of the ISA to understanding the risk at a facility, certain SRP chapters are less dependent on ISA outcomes than others. The chapters concerning radiation safety, environmental protection, emergency management, and decommissioning, for example, contain acceptance criteria that are set primarily by existing regulations and have not been affected by issuing the revision to 10 CFR Part 70. Finally, for new facilities (that have not already been designed, built, licensed, and operated), certain baseline design criteria have been specified in 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities." These criteria identify safety considerations that an applicant must address in its facility design. The ISA for the complete facility design may indicate when reduced levels of assurance may be acceptable.

Each chapter in the SRP includes sections, which are described below in general terms. A more detailed description of the application of the baseline design criteria is given in the discussion of Section 4, "Acceptance Criteria," below.

Section 1., PURPOSE OF REVIEW: This section is a brief statement of the purpose for and objectives of reviewing the subject areas. It emphasizes the staff's evaluation of the ways the applicant can achieve identified performance objectives and ensures through the review that the applicant has used a multi-disciplinary, systems-oriented approach to establishing designs, controls, and procedures within individual technical areas.

Section 2., RESPONSIBILITY FOR REVIEW: This section identifies the organization and individuals by function, within NRC, responsible for evaluating the subject or functional area covered by the SRP. If reviewers with expertise in other areas are to participate in the evaluation, they are identified by function. In general, the Licensing Project Manager has responsibility for the total review product, a safety evaluation report for an application. However, an identified technical specialist should have primary responsibility for a particular review topic, usually an SRP chapter. One or more specialists may have supporting responsibility. In most situations the review is performed by a team of specialist reviewers including the lead reviewer for the ISA and the project manager. Although they individually perform their review tasks, the reviews are extensively coordinated and integrated to ensure consistency in approach and to ensure risk-informed reviews. The project manager oversees and directs the coordination of the reviewers. The reviewers' immediate line management has the responsibility to ensure that an adequate review is performed by qualified reviewers.

Section 3., AREAS OF REVIEW: This section describes the topics, functions, systems, structures, equipment, and components, analyses, data, or other information that should be reviewed as part of that particular subject area of the license application. Because the section identifies information to be reviewed in evaluating the adequacy of the application, it identifies the acceptable content of an applicant's submittal in the areas discussed. The areas of review identified in this section obviate the need for a separate Standard Format and Content Guide.

The topics identified in this section also set the content of the next two sections of the SRP. Both Section 4, "Acceptance Criteria," and Section 5, "Review Procedures," should address, in the same order, the topics set forth in this section as areas to be reviewed. This section also identifies the information needed or the review expected from other NRC individuals to permit the individual charged with primary review responsibility to complete the review.

Section 4., ACCEPTANCE CRITERIA: This section contains a statement of the applicable NRC criteria based on regulatory requirements, and the bases for determining the acceptability of the applicant's commitments relative to the design, programs, or functions within the scope of the particular SRP section. Technical bases consist of specific criteria such as NRC regulations, regulatory guides, NUREG reports, industry codes and standards, and branch technical positions. To the extent practicable, the acceptance criteria identify, as objectively or quantitatively as is feasible, specific criteria and other technical bases that are to be satisfied. The acceptance criteria (including branch technical positions or other information) present positions and approaches that are acceptable to the staff. They are not considered the only acceptable positions or approaches. Others may be proposed by an applicant.

It is NRC's intent that the SRP present acceptance criteria for each technical function area (e.g., nuclear criticality safety, fire safety, radiation safety), and for the management measures (e.g., quality assurance, maintenance, audits and assessments), that allow an applicant to provide a level of protection commensurate with the accident risk inherent in the process activities proposed. For example, at process stations (or for an entire process or sub-process) for which the inherent risk to workers, the public, or the environment is demonstrably small, the applicant needs to provide only those design and operating controls which assure that small risk. The key element in the regulatory transaction involving presentation by an applicant, and review and approval by the NRC, is an adequate demonstration of acceptable control of risk by the applicant, which then supports a competent and informed review by NRC staff. The starting point for the applicant's demonstration of acceptable control of risk is the ISA.

The applicant's ISA summary (described in and reviewed under Chapter 3 of this SRP) is the primary supporting rationale for the safety level of design and operational features. There are, however, design and operational features and management controls that may be required independent of the ISA results presented by an applicant. This is to meet the requirements of 10 CFR 70.64 for new facilities or new processes at existing facilities, or, for all facilities, other NRC requirements such as 10 CFR Parts 20 and 51. The level of detail presented in the ISA summary and in other parts of the application represents the safety basis committed to by the applicant, and it is that basis that is subject to the provisions of 10 CFR Part 70, as revised, regarding changes that a licensee may make to the facility without prior NRC approval. NRC should find an application acceptable if an applicant commits to the design features and management measures defined by the acceptance criteria within this SRP. The criteria in this SRP represent the design features or management measures that support an NRC finding of reasonable assurance of adequate protection, independent of any ISA findings or conclusions that could lead to NRC approval of reduced levels of assurance for certain design features or management measures where the associated risk does not warrant the same high level of assurance.

An applicant for license renewal or an amendment for an existing facility responding to the requirements of 10 CFR Part 70, may propose structures, systems, and components (SSC) or management measures that meet less stringent acceptance criteria than described in the SRP based on supporting analyses from the applicant's ISA. The ISA may be used to justify a reduced level of assurance for particular items relied on for safety, that are associated with lesser risk accident sequences, as defined by the applicant's analysis of likelihood and consequences pursuant to 10 CFR Part 70, as revised. The SRP criteria shown in this SRP apply to those SSC and management measures that are involved in the higher risk accident sequences as defined in Part 70, as revised.

For proposed new facilities or amendments for new processes proposed at existing facilities, the acceptance criteria described in the SRP apply for design purposes and should be addressed in the applicant's licensing submittal for all SSC and management measures and that section's requirement to comply with the baseline design criteria (BDCs) of Part 70, as revised. The BDCs are consistent with risk-informed regulation, in that, for new processes or new facilities, NRC recognizes that good engineering practice dictates certain minimum requirements be applied as design and safety considerations, generally independent of the risk-based information ultimately obtained through the ISA. However, the applicant may use the ISA summary to justify reduced criteria for some SSC and management measures consistent with an ISA summary for a facility final design. Proposed reductions in the level of assurance should be considered by the NRC staff and, if accepted, should also constitute compliance with the BDCs.

Section 5., REVIEW PROCEDURES: This section describes how the review should be performed. It describes procedures that the reviewer should follow to achieve an acceptable scope and depth of review and to obtain reasonable assurance that the applicant has provided appropriate commitments to ensure that it will operate the facility safely. This includes identifying licensee commitments to verify and could include directing the reviewer to coordinate with others having review responsibilities for other portions of the application than that assigned to the reviewer. This section should provide whatever procedural guidance is necessary to evaluate the applicant's level of achievement of the acceptance criteria.

Section 6., EVALUATION FINDINGS: This section presents the type of positive conclusion that is sought for the particular review area to support a decision to grant a license or amendment. The review must be adequate to permit the reviewer to support this conclusion. For each section, a conclusion of this type should be included in the staff's Safety Evaluation Report (SER) in which the staff publishes the results of its review. The SER should also contain a description of the review, including aspects of the review that received special emphasis; matters that were modified by the applicant during the review; matters that require additional information or will be resolved in the future; aspects where the plant's design or the applicant's proposals deviate from the criteria in the SRP; and the bases for any deviations from the SRP or proposed exemptions from the regulations. Staff reviews may be documented in the form of draft SERs that identify open issues requiring resolution before the staff can make a positive finding in favor of the license issuance or amendment.

Section 7., REFERENCES: This section lists references that should be consulted in the review process. However, they may not always be relevant to the review, depending on the action and approaches proposed by the applicant.

2.4 NRC PARTICIPATION IN PHASE IA

NRC staff have participated in various reviews of the TWRS-P program. Under Phase IA, the staff reviewed the following major documents from each of the two contractor teams (BNFL Inc. and Lockheed Martin Advanced Environmental Systems (LMAES); see Chapter 7.0 for main reference for citations):

1. Hazards Analysis Report (HAR) - 1997.

2. Standards Requirements Document (SRD) - 1997.

3. Integrated Safety Management Plan (ISMP) - 1997.

4. Initial Safety Analysis Report (ISAR) - 1998.

Numerous open items or items requiring additional information and clarification were identified by the NRC's review of these documents. Focusing on BNFL Inc. (because of its selection by DOE for continuing into Phase IB - see Section 1.3), a total of 409 comments were generated by the NRC review of these documents, and were subsequently identified and sent to DOE in interagency letters. DOE subsequently forwarded most of these comments to the Contractor for informational purposes. The NRC staff developed and maintained databases to track the issues from the reviews of the SRD and ISAR. The NRC staff identified some 245 comments as being open from the perspective of the NRC as a potential regulator of the facility.

From the point of view of the NRC staff, the most significant issues were found to be the following[19]:

1. Incompleteness of the description and documentation.

2. Key unit operational and process information not included.

3. Inadequate configuration management and documentation organization.

4. Exclusion of safety analyses for the facility workers.

5. Inadequate design class description and SSC categorization.

6. Limited analysis of non-process accidents and external events.

7. Lack of conservatism, particularly at this early stage of design.

The DOE review reached similar conclusions. DOE divided its comments of the ISAR into the following four main categories in March 1998 (see Chapter 7.0, Main Reference 10):

1. Adequate Safety Basis Not Demonstrated: There were 25 issues relating to the adequacy of the proposed facility's safety basis. Resolution was to be required during the first 6 months of Phase IB-1.

2. Inadequate Classification of SSC's: There were seven issues to ensure that the ISMP described in the Contract and in the BNFL Inc. ISMP (plan) were implemented. Resolution was to be required during the first 6 months of Phase IB-1.

3. Incomplete or Conflicting Elements of the Authorization Basis: There were 30 issues that showed incomplete, inadequate assessments and attention to detail. Resolution of these items would be required prior to the submittal of the Construction Authorization Request (CAR).

4. Incomplete Design or Operational Information: There were 28 issues related to the limited information available in the Phase IA (conceptual) design stage. Resolution of these items would be required prior to the submittal of the CAR.

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[19] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, October 17, 1997.

2.5 NRC PARTICIPATION IN PHASE IB-1

DOE selected BNFL Inc. as the sole contractor for Phase IB-1 due to concerns about the status of technology development in the LMAES proposal. Phase IB-1 focused on addressing the main issues from Phase IA and extending the level of design from the preliminary conceptual design level of 3-5 percent to around 25-30 percent. DOE identified the HAR, SRD, and the ISAR submitted in Phase IA as the authorization basis for the continuing design efforts. The NRC staff participated in major reviews of the following (see Chapter 7.0, main reference section for report citations):

1. The Design Safety Features (DSF) submittal (four volumes, Spring, 1999).

2. Topical Meetings and their associated reports (essentially monthly).

3. Other Technical Reports (e.g., the RPP - Radiation Protection Plan, QAPIP - Quality Assurance Program Implementation Plan).

4. Periodic Design Reviews.

5. April 24, 2000, submittal (Firm Fixed Price submittal).

Most of the NRC staff comments on these Part B-1 documents mirror the ISAR comments, and are discussed further in Sections 3.1-3.10 and Appendix A. A large fraction of these comments center around the lack of detail and supporting information on safety-related issues, potentially uneven amounts of redundant equipment and backup systems, and inadequate maintenance of the design/authorization basis. From the perspective of the NRC staff and based upon subsequent NRC reviews of the additional information and documents generated so far in Phase IB-1, a significant number of the original, specific comments (about 100) tracked in the databases remained open. A list of potential critical issues was forwarded to the RU[20]. One large database was subsequently developed for tracking the issues. For this report, these issues have been updated and condensed into a form that are largely design independent and have been placed in Attachment A. DOE opted to discuss several of these issues with the Contractor via the "Topical Meeting" process (see Section 3.1). Most of these meetings had at least one NRC representative in attendance. DOE subsequently decided to close many of these issues by deferring them to the future review of the CAR/Preliminary Safety Analysis Report instead of keeping them open in an Action Plan or SER. DOE/RU anticipates a long list of issues from the CAR review activities[21]. Thus, as of this writing, potentially significant and safety-related issues remain open.

Review areas of major involvement have included the following:

- Organization and general information.

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[20] Leach, M.N., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "Potential Critical Technical Issues For Construction Authorization Resolution," January 21, 2000.

[21] Hoadley, D.A., U.S. Nuclear Regulatory Commission, Note to File, "Minutes of March 16, 2000 Teleconference with DOE/RU," March 21, 2000.

- Seismic and structural considerations.
- Hazards and safety analyses.
- Radiation safety and dose assessment methodology.
- Criticality safety.
- Process and chemical safety.
- Fire protection.
- Explosion protection.
- Environmental protection.
- Quality assurance (QA) and management measures.

These are discussed in more detail in Chapter 3.0. A common concern of the reviews of the NRC staff is the potential lack of conservatism and margin in many of these areas. Thus, if the facility is designed right at a limit in a particular area, it may be significantly restricted for future changes and modifications such as increased capacities or unforeseen changes in the wastes and chemistries. In addition, the design and safety limits are derived from the ISMP. For TWRS-P, ISMP has invoked a circular logic that can magnify the impact of changes, particularly when a scenario consequence result slightly exceeds a limit. ISMP is an iterative process which includes reassessing assumptions (e.g., of radionuclide concentrations) that can reduce a hazard below a limit into a bin with less reliability and safety requirements. Once this occurs, the scenario represents less risk with fewer safety requirements, and, since ISMP focuses on events with higher risk, this scenario and the underlying assumptions may be subject to less scrutiny.

The NRC staff participated with the RU in inspections of Contractor activities and programs. The NRC staff members provided input to the RU inspection teams on the following programs: Personnel Training and Qualification, Design Process, as low as reasonably achievable/RPP, QA, and Employee Concerns. These inspections identified a number of design, QA, and project management issues requiring BNFL Inc. actions, including procedure compliance, design basis documentation, corrective actions, and document control, which are further discussed in the relevant parts of Chapter 3.0.

The NRC staff has also provided input on guidance and regulatory documents to DOE and the RU, and commented on issues that have arisen. Appendix C contains summaries of Point Papers generated by the NRC. Appendix E provides a listing of correspondence with the DOE. Several major guidance and regulatory items reviewed by the NRC staff have included the Limited Construction Authorization Request (LCAR) Review Guidance, the CAR Review Guidance, the co-located worker issue, and risk and regulatory comparisons. The latter two items are discussed further in Chapter 4.0.

The NRC has conducted independent assessments and analyses, including the following:

1. The Standard Review Plan (SRP - NUREG-1702) was developed, issued as a draft for public comment, and issued in final form (see Section 2.3).

2. Point/Position Papers on QA, process safety, fire, and materials aspects (Appendix C).

3. Information papers on radiation protection.

4. CNWRA analyses and reports include:

A. "Survey of Solidification Process Technologies," April 1998.

B. "Process Hazards and Safety Issues for TWRS-P."

C. Vol. 1—Low activity waste Feed Makeup, Solidification, and Offgas Treatment, July 1999.

D. Vol. 2 - Auxiliary Support Systems and Process Control Systems, August 1999.

E. HLW Chemistry Manual.

F. Chemistry of TWRS Waste Pretreatment and Technology Report.

G. PRETREAT Spreadsheet Model for TWRS Pretreatment Flowsheet.

H. Glass Melt Chemistry and Product Qualification Report (in progress).

I. Chemical Simulation of TWRS Pretreatment Process (in progress).

J. "Classification of Process Systems Used in the Tank Waste Remediation System-Privatization Design," June 2000.

The TWRS-P facility represents a radiochemical facility with a relatively large radionuclide inventory. Table 3 compares the potential radionuclide inventories at TWRS-P locations (calculated from Tables 1 and 2) with selected radionuclide quantities at a nuclear power plant (calculated from the Radiological Characteristics Database[22]); the TWRS facility is likely to handle comparable quantities of radioactive cesium, strontium, and technetium in significantly more mobile physical and chemical forms (e.g., as nitrates and aqueous solutions), as compared to ceramic oxide fuels in power reactors. In addition, while a reactor has more energy for potentially energetic scenarios during operations, including scenarios with delays of hours and days before the radionuclide release occurs, the TWRS-P facility has more stored chemical energy for prompt potential events directly involving the radionuclides in their mobile forms. Consequently, the TWRS/Waste Treatment Plant may have some requirements that are more similar to reactor facilities than to commercial fuel fabrication facilities.

[22] U.S. Department of Energy, (U.S.) (DOE). "Radiological Characteristics Database." DOE: Richland, Washington. 1995.

Table 3: Comparison of Curie Quantities between TWRS-P and a Commercial Reactor

Radio-nuclide	Bounding for TWRS-P Facility (curies)					1,000 MWe Nominal Reactor (PWR, in curies)			
						30,000 MWD/MTIHM		60,000 MWD/MTIHM	
	LAW Tanks, 100Kgal (Env. B)	HLW Tanks, 100Kgal	Cs Product, 1Kgal	Cs Resin Column, 100 liters	CST Column	Core, 1 yr cooled	SNF Dry Cask, 5 yr cooled	Core, 1 yr cooled	SNF DRY Cask, 5 yr cooled
Cs-137	8.6E6 (2.3E6)	1.1E6 (7.57E06)	1.32E6 (no change)	72,000 (no change)	3E5 (no change)	9.2E6	1E6	1.8E7	1.9E6
Tc-99	1,000 (727)	1,700 (11,400)	(0)	(0)	0	1,200	140	2,200	260
Sr-90	6,300 (4,500)	1.2E6 (7.57E06)	(0)	(0)	0	6.6E6	7.2E5	1.2E7	1.3E6
TRU	8,700; 70 from solution (8,700; 49 from solution)	17,200 (87,000)	(0)	(0)	0	1.3E7	1.2E6	2E7	1.8E6

Note: Reactor core nominally contains 100 MTIHM and SNF cask nominally contains 12 MTIHM.

PWR values calculated using the Radiological Characteristics Database from Footnote 22.

TWRS-P values calculated from Tables 1 and 2 for the original contract, with 5 percent and 10 percent suspended solids for the LAW and HLW respectively. The values in parentheses are calculated from the new contract, using 2 percent and 20 percent as the suspended solids concentrations for LAW and HLW respectively. The HLW has not been washed. Non-TRU, LAW values do not include the solids contribution; for the new contract, inclusion of the solids contribution would respectively add 7.6E5, 1.1E3, and 7.6E5 curies to the cesium, technetium, and strontium values. Recent discussions have not included CST columns.

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3.0 NRC AREAS OF REVIEW

3.1 ORGANIZATION AND GENERAL INFORMATION

3.1.1 TANK WASTE REMEDIATION SYSTEM-PRIVATIZATION ORGANIZATION

The BNFL Inc. team was approximately 550 people at its peak, with some additional support from outside contractors and consultants. Approximately 50 key personnel from BEI (BNFL Engineering Incorporated, United Kingdom) were in lead positions.

The organization's management functions were adopted from the British Nuclear Fuels Limited parent company. These functions were characterized as the "TWRS-P Business Model." A total of eight key business processes were identified:

Business Process	Responsibility
Management and Administration Corporate policies Strategies Quality Program	Project Manager General Manager Quality Assurance Manager
Resources Human Resources Training	Human Resources Manager
Customer Service DOE Interface and Controls Client (DOE) Commitment Tracking Change Control Stakeholder Interface	General Manager DOE Client Interface
Procurement Local Purchase Plant and Equipment Service Contracts Main Subcontracts (R&D)	Procurement Manager
Finance and Accounts Control of Funds Financial Audit Financial Risk Financial Delegation	Financial Manager
Project Management Schedules Costs Value Engineering Options Project Meeting Structure Project Risk Management	Project Manager
Design and Engineering Engineering Design Process Environmental Safety and Health Procedures and Processes Technical Risk Management Construction	Engineering Manager Environmental Safety and Health Manager
Operations (To be developed)	Operations Manager

Each business process performed in accordance with established methods. The hierarchy of instruction was as follows: general instructions were included in Procedures, with details in Codes of Practice. Other direction was provided by Design Guides, sometimes called Desktop Guides. Procedures, Codes of Practice, and Design Guides were not submitted for U.S. Nuclear Regulatory Commission (NRC) review but, as part of inspection accompaniments with the RU, some Contractor documents and procedures were compared with activities for consistency. Some disconnects were observed, apparently due to the relative fast schedule of the program and the rapid aggregation of the project team from multiple locations.

Performance of some management functions ("business processes") such as Corporate Policy & Strategy, Finance & Accounts, DOE Interface, Schedule & Costs, and Operations were transparent to NRC technical reviewers.

Performance of other management functions was visible to NRC technical reviewers because these departments had active roles in many of the design and safety submittals that were reviewed (Safety Requirements Document [SRD], Hazards Analysis Report [HAR], Integrated Safety Management Plan [ISMP], etc.) and presentations (Design Reviews and Topical Meetings) that were attended. However, it was not clear that environmental, safety, and health (ES&H) and quality assurance (QA) functions were sufficiently independent and had the authority to ensure that their issues were adequately addressed. Specific examples of these points are discussed in Sections 3.1.3 and 3.10.

3.1.2 MEETINGS AND REPORTS ON SAFETY AND REGULATORY ACTIVITIES

The NRC was not fully involved with and did not comment on all meetings and reports. During Phase I, Part A, there were three main deliverable packages for addressing safety and regulatory concerns; these were the Standards Application Package (SAP) (consisting of the SRD and the HAR), the ISMP, and the Initial Safety Analysis Report (ISAR). No revisions were made to these documents and a final version reconciled to the U.S. Department of Energy (DOE) and NRC comments was not issued in Part A. The SRD was revised twice in late 1998 and the HAR was to be revised in the Fall of 2000 (both in Part B-1 of Phase I). In addition, the organization of the documents and references was difficult to follow; for example, the ISAR referred to a Technical Report which in turn referenced three binders of poorly organized, handwritten material. The Part B-1 Contract only identified two main deliverables for regulatory and safety purposes: (1) the Design Safety Features (DSF), also called the six month submittal and (2) the Construction Authorization Request (CAR), which would include the Preliminary Safety Analysis Report (PSAR), at 26 months into the Contract. The NRC and the DOE/Regulatory Unit (RU) also looked at the Firm Fixed Price (FFP), 20-month submittal for informational purposes about the design and approach. The FFP was not intended to address safety and regulatory concerns. The DSF submittal only represented examples that were not part of the design process. Thus, the FFP submittal represented the first formal update of the design and safety features since the ISAR, a period of 2 years. The FFP submittal itself was really a collection of reports. Integration between these multiple reports was not a clear process. Consequently, revisions addressing comments and concerns were not produced in a timely manner, and there were significant time periods between these documents during which there was no contractually required safety information.

As a result of the limited number of safety deliverables in Part B, the Design Review process was instituted by the DOE/RU as part of their approach towards project oversight. The objectives of the design review process were to (a) verify fulfillment of the TWRS-P contract requirement to review the evolving design, (b) ensure that safety-related design aspects were integrated between Contractor engineering disciplines, (c) develop a clear understanding of evolving safety-related design aspects, and (d) provide a reasonable expectation that the CAR would be acceptable. Design reviews were performed primarily as a "status meeting" as the discipline engineers exchanged information, and NRC reviewers attended as many as possible.

DOE and NRC staff members participated in design reviews as observers and not as reviewers. The quality of the NRC-observed design reviews was variable. On the positive side, minutes and action items were kept and assembled for later review in the library. However, on the negative side, greater rigor, meeting focus, and conduct would seem to be needed. Documents were usually provided to the meeting attendees at the beginning of the meeting, which did not allow for adequate review time. The documents/handouts usually consisted of just the overheads for the presentation, without a formal report with revision numbers. No revision changes and written notes/rationale were apparent in the presentations, although these were sometimes mentioned verbally. There appeared to be an emphasis for avoiding action items, using terms such as "design refinement" for items that would seem more like actual action items. It was not clear how action items were tracked, particularly between interrelated design activities and reviews. Although not required, there was little apparent involvement of personnel from the Contractor's safety organizations. The design teams frequently appeared to be working separately from the safety reviewers, without consideration of the design/authorization basis, safety, and the SRD. In some meetings, over half of the questions are asked by DOE/RU and NRC personnel. Finally, design reviews and documentation were not formally reviewed and commented upon by either DOE or the NRC. Only general feedback was provided by DOE to the Contractor.

Topical Meetings were instituted by the RU in Phase IB, in an effort to keep informed of progress and potentially bring closure to the many issues that had been raised during Phase IA. The Topical Meetings were held every month. The results of a Topical Report on the issue were generally submitted in advance of the meeting. After one of the early meetings, an agreement was made to hold a pre-topical meeting approximately 2 weeks prior to the Topical Meeting to ensure that the level of detail in the presentation was satisfactory. An effort was made to have at least two NRC reviewers at each topical meeting. While there were many letter-type Topical and other reports generated for the Topical Meetings, it was not clear how these documents inter-related with each other and would be incorporated into safety analysis reports. A clear route to closure was not apparent. The NRC communicated to the DOE-RU its concerns regarding closure of issues[23]. In particular, the NRC noted the RU was focusing on the Contractor demonstrating an understanding of the standards identification process (see Section 1.4) and the Contractor's implementation of this process, while the NRC concentrated on determining whether or not the Contractor's approach to safety incorporated appropriate levels of conservatism as well as whether the design approaches could later prove to pose potentially unacceptable levels of risk. The NRC further commented that the RU and the Contractor would benefit from closing issues in the near term and well before the CAR; otherwise, the deferrals may increase the time and costs associated with the review of the

[23] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, June 25, 1999.

CAR. The RU identified the following, thirteen significant, unresolved issues from the Topical Meetings[24]:

1. Criticality control.
2. Explosion hazards (hydrogen).
3. Fire protection.
4. Design safety features (unmitigated definition, source terms, uncertainties, etc.).
5. Dose assessment.
6. Emergency response plan.
7. Seismic analysis and criteria.
8. Technology development and plans.
9. Explosion hazards (non-hydrogen).
10. Iodine-129 evaluation.
11. Cesium storage tank cooling.
12. Seismic PRA dose consequence.
13. ISM cycle I.

At the request of the DOE/Office of River Protection (ORP), the RU reviewed the FFP submittal with assistance from NRC staff[25]. While noting the FFP submittal was not intended as a safety deliverable and that some additional design information exists outside of the FFP document, the review determined that the facility and process design information fell far short of the information required for the CAR. Significant additional design information would be needed, including further development of the design and information on safety equipment. The following, short list of items were presented:

1. Limited identification of items relied on for safety (IROFS).

2. Control logic and diagrams not provided.

3. System descriptions were at a summary level, and codes and standards were not identified for most structures, systems, and components, (SSCs).

4. Materials of construction were not identified for most SSCs.

5. Process information focused on normal operations—information for startup, shutdown, offnormal, and fault conditions was not provided.

6. Operating parameters (temperature, flow rates, pressures, etc.) were not routinely included for routine operations and not mentioned for startup, shutdown, offnormal, and fault conditions.
7. Information was rarely provided for system pressure, pressure drops, instantaneous flow rates, and specific gravity (this information would be needed to assess tanks and the adequacy of line/pump sizes, etc.).

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[24] Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc., March 24, 2000.

[25] Gibbs, D.C., U.S. Department of Energy, letter to R. French, U.S. Department of Energy, May 25, 2000.

8. The description focuses on operability with little attention to safety issues.

9. Information regarding analysis and the validation/verification of software for structural and seismic considerations was not provided.

10. Important-to-safety (ITS) areas of the design are not complete or do not have the detail to support safety analyses (CAR/PSAR).

11. The levels of conservatism and the design margins are not mentioned for operations or safety parameters.

The RU subsequently provided this information to the Contractor and included a more detailed report by review area[26].

3.1.3 INSPECTIONS

NRC reviewers also participated in several inspections which were conducted by the DOE/RU: these included inspections of BNFL Inc.'s Employee Concerns Program, Training and Qualification Program (two inspections), Quality Assurance Program (two inspections), and Design Process.

One of the earliest inspections was of the Employee Concerns Program. Inspectors found the program to be in place, but no formal concerns had been voiced.

An inspection of the Contractor's Training and Qualification Program was conducted early in the project. The Contractor was hiring staff at the time of the inspection. Training and Qualification procedures were only recently issued, but appeared to be functioning. It was determined that the Contractor's staff was qualified for their positions and responsibilities. Two important program weaknesses were identified: (1) the positions of Training Manager and Training Specialist were not filled at the time of the inspection; and (2) the Contractor's program for position-specific training was not fully developed. The second inspection of the Training and Qualification program confirmed that deficiencies in management and position-specific training still remained.

An inspection of the design process and authorization basis was initially scheduled in the Spring of 1999, then postponed by the RU because of concerns that the Contractor would do poorly as the authorization basis was not being followed nor maintained. Subsequently, the postponed inspection was canceled - again, over RU concerns that the Contractor would not pass the inspection. The Contractor design process was inspected in January, 2000. Three (3) areas of Findings were identified during the inspection: (1) examples where the Contractor had not followed procedures; (2) procedures did not address or consider testability or inspectability in the design; (3) the Quality Assurance organization was not effectively reviewing the design program. Limited QA overview of the design process and continued instances of procedural noncompliance were causes for concern. These and continuing concerns about the management of the authorization and design basis resulted in a Corrective Action Notice (CAN) and conference in March, 2000. A highly interactive QA organization that self-identified

[26] Gibbs, D.C., U.S. Department of Energy, letter to P. Strawbridge, BNFL Inc., June 2, 2000.

problems, such as the procedure noncompliance issues identified in this inspection, coupled with a robust corrective action program, were two elements required to assure the development of a safe design and maintain the authorization basis.

DOE/RU and the Contractor recognized deficiencies in QA, and conducted several self-assessments, including a Root Cause Analysis. This was viewed as a weakness in Project/Management, and not simply a QA problem. Some of the deficiencies include a failure to follow procedures, lack of knowledge of requirements, failure to implement inspection programs, inadequate documentation of design activities, and deficient design or contract technical and standards bases. The failures noted were program/project implementation (management) problems, which would be detected during inspections.

DOE/RU issued a CAN[27] on March 14, 2000, and held a public meeting in March 2000. Contractual and regulatory documents specify that changes in facility design will either be consistent with the existing authorization basis or the authorization basis will be revised based upon a determination of the potential impacts of the proposed changes. The RU determined that, to the contrary, the Contractor had neither established nor implemented a program or process to ensure that the authorization basis was maintained current with respect to the facility design. In short, the facility design was proceeding without consideration of the authorization basis and safety issues. This condition existed throughout Part B1 of the TWRS-P project, from contract award (August 1998) through two inspections (October and November 1999) through several meetings with the Contractor until the CAN was issued. The RU and the Contractor held several other meetings and the RU ultimately accepted the Corrective Action Plan (CAP)[28]. The RU had also planned an inspection on the self-assessment and the CAP[29]. However, the contract termination resulted in the indefinite rescheduling of the inspection and CAP[30].

The DOE/RU regulatory program has a significant amount of flexibility and discretion for regulatory decisions and actions. This allowed the postponement and subsequent cancellation of an inspection that would have resulted in a corrective action notice a year earlier than when it actually was issued, and, as a result, most of the design activities occurred outside of the authorization basis and SRD. The RU subsequently received a large number of authorization basis amendment requests (ABARs) in the last 3 months of the contract. The CAN was not issued until 1 month before the FFP. If the CAN had been issued a year earlier, there would be more confidence that the design is within the authorization basis. In the NRC regulatory regime, it is unlikely that such an inspection would have been postponed and then canceled if the licensee was known to be outside their design basis.

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27 Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc., March 14, 2000.

28 Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc., April 25, 2000.

29 Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc. March 27, 2000.

30 Gibbs, D.C., U.S. Department of Energy, letter to P. Strawbridge, BNFL Inc., July 26, 2000.

3.1.4 SUMMARY

The Contractor organization experienced rapid growth during Phase IA and Phase IB-1. Adherence to procedures and protocols, and integration of program activities and documentation (authorization basis) were not fully developed. The emphasis on schedule and cost may be contributing to these concerns. In the near term, this is unlikely to change because of new contracts and new contractors.

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3.0 NRC AREAS OF REVIEW

3.2 SEISMIC AND STRUCTURAL CONSIDERATIONS

3.2.1 TANK WASTE REMEDIATION SYSTEM-PRIVATIZATION: PART A

The privatization concept developed by the U.S. Department of Energy (DOE) for the Tank Waste Remediation System-Privatization (TWRS-P) project set forth the contractual requirements for the two contractors in Part A, and was expanded on in a series of four top-level documents. These were known as DOE/RL-96-003, -004, -005 and -006. Briefly these documents addressed the following topics for the two contractors, LMAES and BNFL Inc.[31]

-003: the DOE Regulatory Process for TWRS-P
-004: Process for Establishing a Set of Standards and Requests
-005: Concept of the Regulatory Process for TWRS
-006: Top-Level Principles

Based upon these documents, it was the responsibility of each contractor to devise a pretreatment process, a vitrification process and all the auxiliary and supporting processes, and the plant design so that the treatment of the tank wastes would result in a product that will meet the acceptance criteria for high level waste disposal. The hazards specific to each of the contractors' processing activities were to be identified and evaluated for their impact on the safety on each of their proposed facilities throughout its lifetime. The DOE privatization concept included the execution of an integrated safety management system that would consider the hazards, the consequences of failures, and the mitigation elements that would be provided in the facility design and operation. Each contractor, based on implementing these four top-level documents, was to identify and recommend for DOE approval, a set of safety standards that the contractors would certify that, if properly implemented, would result in adequate safety, compliance with applicable laws and legal requirements, and conformance to the DOE-stipulated top-level safety standards and principles. For the NRC, the elements of the defined regulatory process with the most importance to the NRC at this stage of the project were the Standards Approval, the Initial Safety Evaluation, and the Construction Authorization Request (CAR). These elements in a final form would reflect similar aspects that the NRC would consider in a preliminary safety analysis report (PSAR) that would be submitted by an applicant for a license prior to NRC authorizing construction of a facility. In this instance however, DOE was interacting with the contractor (or applicant) in the conceptual design and preliminary design phases. These represent a much earlier stage of involvement than the NRC would normally be engaged in. DOE, in this situation as being responsible for the cost of the facility as well as safety, was motivated to be involved in the entire developmental process for this one-of-a-kind facility.

The Standards Approval was a step in the Part A contract procedures to allow for DOE review, evaluation and subsequent approval, if warranted, of each of the contractors' proposed set of radiological, nuclear, and process safety standards and requirements, as each contractor defined in their Safety Requirements Document (SRD). In addition, each contractor was to

[31] Note that while there were two contractors involved in the work under Part A, all specific comments relative to this part are based only on the submittal by BNFL Inc. since the other contractor LMAES, did not proceed to Part B.

provide a standards-based integrated safety management program that was to be documented in an Integrated Safety Management Plan (ISMP). DOE required that the SRD document all requirements of applicable laws and regulations, conform to the -0006 top-level document, and that the set of standards and requirements identified by a contractor was generated by the standards process defined in the -0004 top-level DOE document using the appropriate level of expertise. The SRD was also to reflect the fact that all hazards associated with the proposed facility and its operation have been appropriately addressed. The ISMP, developed as a portion of the Standards Approval process, was specified by DOE to include the planning elements of the implementation plans required by DOE regulations, particularly 10 CFR Part 830 addressing Nuclear Safety Management. DOE required that the contractor's selected safety management process documented in the ISMP be standards and requirements base, and that it be appropriately tailored to the hazards associated with the contractor's proposed facility, its operation and its deactivation.

The only specific guidance provided by DOE that could be considered as a yardstick against which the level of safety of each of the proposed facilities could be judged consisted of the radiological Dose Standards Above Normal Background for individuals. These consisted of a breakdown for individuals in various categories, under various event scenarios, with varying estimated probabilities of occurrence that were provided in terms of numerical dose values, except for two types of individuals (the worker and co-located worker) impacted by the unlikely events and the highly unlikely events. The dose limits for these two cases were to be provided by the contractors as reflected by their proposed design. These standards were provided by DOE in the -0006 top-level document with the statement that the use of the standards does not provide a blanket waiver to safety regulations that apply to DOE activities, but are additional considerations that must be addressed by each of the contractors.

The BNFL Inc. initial SRD submittal (September 26, 1997) that was reviewed identified two types of standards along with the source of the standards as follows. Standards used to specify certain objectives that must be achieved, or in other words performance objectives, were identified as "safety criteria." It was noted that these standards usually are made up of laws and regulation requirements. Standards that are a prescriptive method for achieving a certain objective were identified as those standards that are industry consensus codes or standards such as American Society for Testing and Materials standards. The initial submittal also indicated that the consequential hazards for the facility consisted of those that were chemical and/or radiological in nature. It was also indicated that the hazard analysis would identify the need for accident prevention controls or mitigation controls in order to meet the performance goals. The mitigation controls were noted as being either engineered or administrative with a preference for engineered mitigation features.

With respect to the engineering and design of these mitigation features, BNFL Inc., as a result of the process hazards analysis, elected to identify two classes of structures, systems and components - Design Class 1 and Design Class II.

Design Class I was designated for those structures, systems, and components (SSCs) where structural controls were indicated from the preliminary accident analyses as being necessary to protect the public from seismic-induced failure of the high level waste (HLW) receipt tanks.

The Design Class II SSC category based on the preliminary accident analyses indicated that the workers needed to be protected from the following list of accidents:

1. Failure of the HLW feed receipt tank.
2. Self-boiling of the cesium product.
3. Cesium ion exchange column fire.
4. Melter failure.
5. Cesium product storage overpressure or canister drop.

BNFL Inc. noted that engineered features required to prevent or mitigate these accidents and the resulting hazards would be provided in the facility. The engineered features were envisioned to consist of those elements providing confinement for the process cells, the ventilation system for the process cells and the cell ventilation stack. Other potential Design Class II elements included certain radiation monitors, instrumentation and controls including interlock logic devices controlling entry into high radiation areas, isolation devices relied on for maintaining ventilation confinement control of radioactive material, and shielding in certain areas. This information was presented by BNFL Inc. as constituting the "safety criteria" standards.

The consensus codes and standards were the next portions of the Standard Approval process and provided a more usable set of standards for the seismic, civil and structure technical discipline subject area. BNFL Inc. stated that the intent of the design process was to use consensus codes and standards that are recognized, and accepted. Design documents would be based on DOE guidance, and US and United Kingdom commercial nuclear and chemical industry practices for designing SSCs. In cases where no consensus code or standard exists, BNFL Inc. indicated an ad hoc standard will be developed. At the time of the submittal of the initial SRD, the need for any such ad hoc standards had not been identified. For the seismic, civil and structural discipline areas, BNFL Inc. relied heavily on the consensus codes and standards used in the commercial US nuclear power industry along with standards and guides that have been developed and incorporated into the DOE organizational hierarchy. In addition, guidance documents used by the U.S. Nuclear Regulatory Commission (NRC) in the licensing of commercial power reactors were also listed within the SRD.

Design loads resulting from natural phenomena were provided in tabular forms for each design class, i.e. for Design Class I and Design Class II SSCs. The main document referenced in these tables was DOE-STD-1020-94 with Change No. 1, 1996, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities." Additionally, DOE-STD 1021-93 with Change No. 1, 1996, "Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems and Components" was referenced, and for straight wind loads and snow loads the references were to ASCE 7-95, "Minimum Design Loads for Buildings and Other Structures," American Society of Civil Engineers. Tornado loads and tornado missiles were noted as not applicable for the Hanford Site for any design class of structure, however a wind generated missile was to be considered for Design Class I structures. Subordinate to these referenced documents were the implementing codes and standards that were identified for both design classes. The basis for the seismic loads for Design Class I was noted to be a previous report for the Hanford site that had been encompassed for application to all facilities at the Hanford site. This was known as the "Geomatrix Report (1996)" which had the title of, "Probabilistic Seismic Hazard Analysis DOE Hanford Site, Washington," WHC-SD-W236A-T1-002, Rev. 1. Based on this document BNFL, Inc. indicated that an equal-hazards response spectra for a peak horizontal ground acceleration of 0.24g at 33 Hz and a peak vertical ground acceleration of 0.16g at 50 Hz was appropriate for the facility. A graphical representation of these spectra was provided for the case of 5 percent damping. The codes and standards referenced for Design Class I included ASCE 4-86, "Seismic Analysis of Safety-Related Nuclear Structures;" ACI 349-90, "Code Requirements for Nuclear Safety-Related Concrete Structures;" AISC N690-95, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities;" and UBC-94, "Uniform Building Code." The basis for the seismic loads for Design Class II was identified as the Uniform Building Code, UBC-94, for Zone 2B and an Importance Factor of 1.0. The reinforced concrete design was to follow ACI 318-95, "Building Code Requirements for Structural Concrete," and the structural steel design was to follow AISC-89(ASD), "Specification for Structural Steel Buildings-Allowable Stress Design(ASD)."

NRC comments in this technical area were provided to the DOE/Regulatory Unit (RU) as part of the NRC Comments on the BNFL Inc. Standards Approval Package, transmitted on 10/17/97 and 10/30/97. The following items in the category of hazards and codes/standards were addressed and identified as requiring additional information:

1. Natural events and the related hazards that are beyond the design basis should be included in the hazards analysis and the appropriate mitigative measures identified as needed should be identified and discussed.

2. Basis of probability of occurrence for design precipitation intensities relative to the length of meteorological data at the Hanford Site needs to be provided.

3. The Hazards Analysis Report (HAR) indicated that tornados would not be considered based on the low annual probability of occurrence (once every 100,000 years) and the low wind speed that would be less than the straight wind speeds. This event is still categorized as a Frequency 1 event per Table 3-3 of the HAR. In addition, besides the wind velocities associated with a tornado, there are rotational velocities and differential pressures that can be potentially dangerous for closed structures and those with controlled pressure differentials for contamination control. The basis for ignoring all effects of tornados, even though the tornados may be less intense than a maximum tornado, should be justified further.

4. The HAR indicates that the return period for the design basis earthquake is approximately 2000 years. The basis for the peak horizontal ground acceleration being 0.24 g for this return period needs to be provided. Based on the DOE top-level documents this is an unlikely event. There should be information provided regarding the consideration and treatment of seismic events in the extremely unlikely category of seismic events which have a return period from10,000 to 1,000,000 years.

5. The SRD was noted to reference numerous commercial nuclear power plant standards, portions of which may be inappropriate, or conflicting for TWRS-P application. The all-encompassing references to such standards should be re-examined and the specifics identified.

6. The use of non-US consensus codes and standards should be compared to the associated US codes and standards. The development, adoption, and use of ad hoc standards should be clearly prescribed in a documented procedure.

7. References in several locations in the document to the Uniform Building Codes (UBC) does not identify the year of the UBC, or identifies an old version of the UBC. These issues should be clarified with references corrected to the current UBC edition.

8. The safety classification system of the SSCs into the "design classes" relative to the hazards and the consequences does not appear to be consistent.

As noted by the RU in a letter to NRC, dated 12/5/97, these items were already addressed by RU questions to BNFL Inc., or provided information for additional questions the RU asked of BNFL Inc. The RU considered Item 6 to be editorial, but NRC disagreed with that position. While BNFL Inc. provided the RU additional information in response to some of the above issues as well as others, the RU's Evaluation Report of the BNFL Inc. SRD (RL/REG-98-01, March 1998) identified issues in this technical discipline area requiring resolution prior to commencement of preliminary design as well as others requiring resolution prior to construction authorization. With respect to hazards, the top-level Safety Criteria 4.1-3, 4.1-4 and 4.3-3 were identified as only addressing the effect of natural phenomena and hazards, but not all categories of potential hazards. The evaluation identified that Safety Criteria 4.1-3 and 4.1-4 also established seismic design criteria for which BNFL Inc. had not provided an adequate safety basis, and the criteria only addressed Design Class I and Design Class II SSCs and not all SSCs important to safety. The design classification system as proposed by BNFL, Inc. was identified as being inadequate and requiring changes that BNFL Inc. had described in a letter, dated 2/19/98. Correction of these deficiencies was required by the RU prior to the commencement of preliminary design. With regard to the site description and its impact on design, the RU's evaluation stated that adequate justification was needed to support the BNFL, Inc. hazards approach for natural phenomena hazards design criteria, and this should be more than that described in the HAR. The RU stated that BNFL Inc. did not provide an adequate or sufficiently detailed safety basis to support the selection of a 0.24 g horizontal acceleration earthquake with a 2000-year return period. The RU noted that these were issues that are commonly resolved during preliminary design, but essential to resolve prior to construction; hence, it was required that these be resolved prior to construction authorization. With respect to codes/standards, the RU determined that the implementing codes and standards were adequate as subordinate standards. The NRC had noted that two of the referenced standards were currently outdated by a later edition and recommended that ACI 349-97 and UBC-97 be the referenced standards. This was not reflected in the RU safety evaluation report.

Another key element within Part A of the DOE contract was the development, by each of the contractors, of an Initial Safety Assessment (ISA) for the proposed TWRS-P facility. DOE, as part of the contract with its contractors, required that the ISA provide the following elements (the NRC also believed these elements to be relevant for its technical safety review and support to the DOE RU):

1. An adequate description of the site, the design of the proposed facility and the operation.

2. A description of the hazards, the potential design-basis events and the analysis of those potential events.

3. Preliminary acceptance criteria.

4. Identification and description of SSCs important to safety and the basis for the classification.

5. Evaluations of the proposed design for constructability, operability, reliability, maintainability and inspectability.

6. An Initial Safety Analysis Report (ISAR) that defines the projected safety basis for the facility in terms of physical design, SSCs with prescribed safety functions, the operating modes and conditions, along with representative events of off-normal internal events, and external events. The results of representative safety analyses considering uncertainties should describe how the facility will meet the SRD requirements as well as how the prescribed dose limits are met.

The BNFL Inc. ISA submittal (January 12, 1998) was reviewed in the technical area of seismic, civil, and structural areas, and a series of comments/questions were developed. In some cases the subject of the issue represented a broader focus that these technical disciplines whereas in other cases the issue was unique to these disciplines. NRC comments in this technical area were provided to the DOE RU in a letter, dated 2/6/98, in support of their review process. A total of ten comments/questions resulted from the NRC review in these particular technical disciplines. These can be broadly identified as issues related to the design safety classes/categories, to external events (both natural and man-induced), and to the availability of information on various buildings and their function in the complex and the hazards they may pose. The consolidated issues addressed the following topics

1. Three design classes are noted in the ISAR with Design Class I and II being for the protection of the public and workers respectively, however all the credible scenarios which can change the required safety classification do not appear to have been considered.

2. All potential events considered are not identified in the ISAR and those considered incredible are not identified nor is the basis for such a classification provided.

3. Design Class II SSCs appear to be treated differently in the SRD and the ISAR relative to external events.

4. Other adjacent sites and facilities hazards are not always considered.

5. Specific design criteria were questioned regarding the basis of the criteria.

As noted by the RU in a letter to the NRC, dated 2/26/98, the comments were already encompassed by RU comments/questions to BNFL Inc. or revisions were being made to those items, or a new comment/question was being sent to BNFL Inc. The RU noted that three of the original NRC comments were determined to not address information that DOE believed was needed at this stage of the design development process, so they were not considered further. The RU Initial Safety Evaluation Report of the BNFL Inc. ISA (Main Reference 10) identified issues in this technical discipline as requiring resolution. The issues for the entire facility as a result of the ISA were grouped into four categories as follows: (1) adequate safety basis not demonstrated, (2) inadequate classification of SSCs, (3) incomplete or conflicting elements of the authorization basis, and (4) incomplete design or operational information. The schedule for resolution was identified in terms of three distinct time frames as follows: during the first six months of the preliminary design (Part B-1), during the PSAR stage, and prior to the submittal of the Construction Authorization Request.

The RU stated that the justification for the use as a design basis earthquake of an event with a peak horizontal ground acceleration of 0.24g, with a return period of 2000 years, had not been derived or provided based on a comprehensive hazard/consequence analysis. Further, it was noted that there was a need to identify and justify how natural phenomena hazard design requirements apply to SSCs and their safety functions. The use of specific editions of codes and standards as well as specific provisions were identified by the RU regarding tornados, site precipitation flooding, and assigned importance factors. The lack of any site-specific geotechnical investigation results was also identified and there was a stated need for the facility structural classifications to be defined. The RU identified these items for resolution within 6 months after the preliminary design is initiated.

As a result of work underway in the NRC in 1998 to revise 10 CFR Part 70, which was ultimately expected to be the regulation the TWRS-P facility would have to comply with if the private owner-operator were to be licensed by NRC, an Issue Paper was developed and provided to the RU. The paper, "Consideration of the Seismic Events for Integrated Seismic Analysis of Hanford Tank Waste Remediation System," dated 6/29/98, discussed recently issued consensus standards and guidance that could be considered as the baseline minimum set of seismic criteria.

3.2.2. TANK WASTE REMEDIATION SYSTEM-PRIVATIZATION: PART B-1

When Part B-1 of the contract between DOE and BNFL Inc. began, revisions were being made by BNFL Inc. to the previous documents in response to the issues identified. A series of topics that identified specific high-interest issues judged to be key issues needing technical resolution were scheduled. In this discipline area, the identified topic was "seismic;" however, before the first scheduled Topical Meeting was held there were several pre-meetings. NRC staff attended such a meeting on 11/6/98 where the topic of discussion focused on BNFL Inc.'s request to obtain concurrence from the RU on the peak ground acceleration value for the seismic analyses of the required SSCs. The basis for the BNFL, Inc. proposed design basis earthquake of 0.24g peak horizontal ground acceleration was identified as DOE-STD-1020-94, with Change No. 1, 1996, and a report addressing the entire Hanford site entitled, "Probabilistic Seismic Hazard Analysis, DOE Hanford Site, Washington," WHC-SD-W236A-TI-002, 2/14/94 through Rev. 1A, 10/8/96, 02, by Geomatrix Consultants, Inc. Issues that were identified included the fact that it was not clear that new data and new concepts on the seismicity of the area may be available since the time the Geomatrix report was developed, whether distant events could be more dominant in some frequencies than nearby events, the impact on response of reflected energies from a bowl/basin effect for the Hanford site, the extent of the peer reviews conducted on the Geomatrix report, and the level of BNFL Inc. review/validation of the report to conclude that it can be applied to the TWRS-P facility. As a result of meeting on 11/6/98, BNFL Inc. submitted additional information on 11/19/98 for use in the Seismic Topical pre-meeting that was held on 12/6/98. NRC provided comments to the RU (12/11/98) after a review of the additional BNFL Inc. information. The key issues that had been identified included the following:

1. The submittal is a much broader request for approvals than just the horizontal peak ground acceleration value. It also included the response spectra and a document outlining the seismic design approach.

2. Noted that the submittal seemed to contain conflicting information regarding the Design Basis Earthquake and the Performance Categories of DOE-STD-1020-94, and the return period of 2,000 years.

3. While mentioned in the submittal, it was unclear how the seismic analysis and design of SSCs in other than Performance Category 3 were to be addressed.

On December 14, 1998, a Topical Meeting was held on the subject of the TWRS-P peak ground acceleration in order to discuss the basis for the derived value. However because the pre-meeting discussion had included related topics, the RU asked BNFL Inc. to address the relationship of the 2000 year return design basis earthquake to the extremely unlikely events and the consequences that must be met for the project when considering the Radiation Exposure Standards for Workers (i.e., these must be addressed for events with return periods from 10,000 to 1,000,000 years). As a result of the meeting there were several key items left for resolution:

1. The current validity of the Geomatrix Report for use on TWRS-P must be developed including addressing basin effects, site specific attenuation data, uncertainty, and peer review.

2. The use of the DOE series of standards related to DOE-STD-1020-94.

3. The dip in certain frequencies of the proposed horizontal ground response spectra for 0.24 g; d.) The basis for the statement that there was a 5 percent chance of building collapse for a beyond Design Basis Earthquake.

Target dates for additional information were established by the RU and BNFL Inc. for February 1999 for all items except a May 1999 date was set for the availability of a document entitled, "Seismic Design Approach/Criteria." NRC comments on the Topical Meeting were contained in an NRC Memorandum dated 2/2/99 that reiterated these major outstanding issues.

A meeting was held on January 7, 1999 as a follow-up meeting to the December 14, 1998 Seismic Topical Meeting to focus on the Geomatrix Report and the questions that arose from the review of the report by the RU and its consultants. A list of questions the RU had developed prior to the Seismic Topical Meeting that remained unanswered as well as additional questions developed from the Seismic Topical Meeting were the focus of this meeting. The more important issues included the following:

1. Relationship of geologic structure to the historical seismicity.

2. Comparison of attenuation relationships used for subduction-zone earthquakes to the empirical database in the range of distances to the site.

3. Method of scaling peak accelerations for subduction-zone earthquakes for the site-specific response analyses.

4. Comparison of input parameters for the probabilistic seismic hazard analysis to the US Geological Survey (Frankel, 1996) values.

Most of the issues were satisfactorily resolved in the meeting. However there was a list of nine additional issues/questions that remained. These were provided to BNFL Inc. by the RU in the meeting minutes issued, dated 2/1/99. At the meeting it was agreed that BNFL Inc. would provide the answers to the additional issues on 2/28/99 in a report. The key items in this list included the following: justification for the weighting factors used for the various fault models in the Preliminary Seismic Hazard Analysis (PSHA) logic tree relative to the geological characteristics should be provided and there should be some sensitivity calculations to demonstrate the impact of weighting factors on the peak ground acceleration value as well as the response spectra.

On March 18, 1999, BNFL Inc. transmitted a package of seismic documents to the RU with the statement that the four enclosures adequately addressed all the comments and questions raised by the RU and provided an acceptable basis for the design basis earthquake being a peak horizontal ground acceleration of 0.26g. A request for approval of that value was requested. The four enclosures were as follows:

1. "TWRS-P Facility Design Basis Earthquake-Peak Ground Acceleration, Seismic Response Spectra, and Seismic Design Approach," Rev. 1, 3/17/99.

2. "Applicability of DOE Documents to the Design of TWRS-P Facility for Natural Phenomena Hazards," Rev. 0, 3/17/99.

3. "Validation of the Geomatrix Hanford Seismic Report for Use on the TWRS Privatization Project," Rev. 0, 3/17/99.

4. "Approach for Ensuring Compliance with the TWRS-P Radiation Exposure Standards under Earthquake Conditions," Rev. 0, 3/17/99.

The NRC reviewed these documents and provided comments to the RU in a letter dated 5/10/99. The comments consisted of 10 items related to Enclosures 2 and 4. NRC summarized the comments in the transmittal letter by stating the following, "...whether or not the adoption of a 2,000 year return period for the seismic design basis event will result in a TWRS-P facility that will meet the DOE radiation exposure standards for this project has yet to be demonstrated." Additionally it was stated that, "...while Enclosure 4 to BNFL Inc.'s submittal presents and proposes the use of a methodology that might be applicable for such a demonstration, the details for its execution and the source for the needed data to conduct such an analysis have yet to be determined. Consequently, the possibility remains that some SSCs might have to be designed to a seismic level (i.e., peak ground acceleration) above 0.26g, or that changes in the stress allowables or load factors might have to be made to provide reasonable assurances that allowable exposure levels (dose limits) will not be exceeded with a frequency of occurrence per year equal to or greater than one in one million."

The RU in a May 14, 1999, letter to BNFL Inc. described a conditional acceptance of the peak horizontal ground acceleration at 0.26g that stipulated that their enclosed concerns must be satisfactorily resolved.

BNFL Inc. presented a seismic probabilistic risk analysis (PRA) approach at the Topical Meeting in October 1999. This was comprised of several models on different failure modes and releases. Extensive use was made of fragility curves and "small leak before break" arguments, for both civil structures (e.g., cells) and equipment (e.g., tanks). The median dose result (50 percent) for collocated workers corresponded to the SRD limit of 25 rem with a frequency of 1E-6/yr. However, the analysis indicated the facility would survive and have functioning prevention and mitigation features with earthquakes considerably stronger than 0.26 g. Upon questioning, BNFL Inc. indicated that input values had been assumed to illustrate and refine the methodology. Actual data and design values would be used after several months of further model development and with actual design parameters and features (i.e., once finalized - after the Firm Fixed Price submittal).

3.2.3 PRESENT STATUS

At the time of the contract cancellation by DOE, the issue related to the seismic area of review was the acceptability of the exact PRA methodology that would be used and the validity of the input data, in order to substantiate that the dose limits for the extremely unlikely events will not be exceeded. The criteria and methodology that will be used for the civil and structural design are outlined in the project documents and are considered to be acceptable once there is agreement on the safety classifications of SSCs. The site specific geotechnical data were provided in the last BNFL Inc. submittal prior to contract cancellation; however the NRC was not provided a copy of that specific material. The NRC has no information regarding DOE acceptance of that geotechnical information and its impact on the soil and structural foundation designs. In addition, the NRC does not have any information on the compatibility of current and near-term site activities with the proposed design or plans from the new, Management & Operations contracts.

3.0 NRC AREAS OF REVIEW

3.3 HAZARDS AND SAFETY ANALYSES (ISA/ISM)

3.3.1 STANDARDS APPROVAL PACKAGE (SAFETY REQUIREMENTS DOCUMENT AND HAZARDS ANALYSIS REPORT) AND INITIAL SAFETY ANALYSIS REPORT

The Safety Requirements Document (SRD) and Hazards Analysis Report (HAR) arrived from BNFL Inc. in September 1997 and identified numerous process and chemical safety scenarios (many with both radiological and chemical consequences) using a preliminary hazards-operability (HAZOP) method. HAZOP reviewers with expertise in similar types of plants and processes were used to bridge areas of incomplete information because the preliminary design was still undergoing development and, as such, many open areas existed. The HAZOP used an American Institute of Chemical Engineers (AIChE) risk matrix (a binning approach) to assist with identifying those potential scenarios that posed a sufficient risk (i.e., consideration of both frequency and consequence) to warrant quantitative risk analysis and the consideration of safety controls. In general, the list included many entries of a benign nature that obscured some of the more hazardous conditions at the proposed facility. In addition, the expert review panel inadvertently included the effects of mitigation in some of their assessment, and, consequently, the frequency and consequence bins for some events were underestimated. Thus, potential items relied on for safety (IROFS) were overlooked. From these reviews, the following comments were raised in the hazards and safety analysis area:

1. Incomplete process and hazard descriptions.

2. Inconsistency of unmitigated and mitigated analyses.

3. Emphasis on active instead of passive mitigation and control.

4. Organization of documents and information.

5. Inconsistencies in hazards identification, structures, systems, and components (SSC) categorization, and design classes.

6. Relative lack of quantification.

7. Presence of large tanks and inventories.

8. Limited consideration of offnormal and unanticipated events, and interactions.

The actual comments may be found in the staff's SRD/HAR comment transmittal letter[32].

BNFL Inc.'s Initial Safety Analysis Report (ISAR) was dated January 12, 1998. The ISAR represented a more detailed design as compared to the SRD/HAR package. The ISAR advanced the design and presented quantitative analyses on several process and chemical safety scenarios, including tank failures, crystalline silicotitanate (CST) powder dispersion, ion

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[32] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, October 17, 1997.

exchange resin fires, glass spill, anhydrous ammonia leaks, and a nitric acid spill. However, the level of the design corresponded to a preliminary, conceptual level, identified as circa "2-3% of design" by the U.S. Department of Energy (DOE) and the Contractor. From the perspective of safety reviews and regulation, the ISAR did not contain sufficient information for addressing many of the concerns and issues. Furthermore, the input values and methods used appeared to be nonrepresentative and would underestimate the potential consequences. From the ISAR review, the NRC staff raised the following comments and questions in the process and chemical safety arena:

1. Lack of conservatism.
2. Inconsistency of unmitigated and mitigated analyses.
3. Emphasis on active instead of passive mitigation and control.
4. Organization of documents and information.
5. Inconsistencies in hazards identification, SSC categorization, and design classes.
6. Relative lack of quantification.
7. Presence of large tanks and inventories.
8. Limited consideration of offnormal and unanticipated events, and interactions.

The actual, detailed comments may be found in the NRC staff's ISAR comment transmittal letter[33].

3.3.2 DESIGN SAFETY FEATURES SUBMITTAL

The Design Safety Features (DSF) submittal from BNFL Inc. consisted of two parts. Part 1 discussed general features used to address general hazards and concerns. In essence, this part presented the practical interpretation of the codes and standards from the SRD. "Standard" chemical process industry approaches would be used unless radiological limits were shown to be exceeded. Part 2 discussed and analyzed 10 examples in more detail, including loss of confinement events with both radiological and chemical consequences. The 10 examples were:

1. Hydrogen generation in the high level waste storage vessels.
2. Loss of cooling to the cesium storage vessel.
3. Load drop of a pretreatment pump (out of cell).
4. High level waste melter feed line failure.
5. Cooling water contamination.
6. Sample carrier breakout.
7. Low activity waste pipe break.
8. Receipt vessel rupture.
9. Activity backflow.
10. Nitric acid handling.

The following, main comments and questions were raised by the NRC staff regarding safety analyses from the DSF review:

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33 Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, February 6, 1998.

1. Concerns about adequate margin, conservatism, and defense-in-depth resulting in potential under-estimation of consequences and IROFSs.

2. Relative lack of information, details, specificity, and numerical values on many features.

3. Design basis not identified vis-a-vis "best basis."

4. Uneven approach to dose methodology including the handling of unmitigated events.

5. Use of optimistic design assumptions and reliabilities.

6. Uncertainties and inconsistencies not addressed.

7. Validation and justification of data, assumption, and models.

The actual comments may be found in the NRC staff's DSF comment transmittal letter[34]. The DSF constituted an example of approaches for safety analyses and did not represent an actual design or safety document.

3.3.3 RELATED ACTIVITIES AND TOPICAL MEETINGS

The DSF submittal and related subjects were discussed at the Topical Meeting that was held in Hanford on March 27, 1999. The lack of detail was noted on the design and many of the proposed safety features. The April 1999 Topical Meeting presented the dose methodology to be used in the Contractor's integrated safety management (ISM) process. Many questions were raised and there were concerns about the limited documentation, the relevance of the Sellafield database, source term calculations, and the incorporation of decontamination factors in unmitigated analyses. Subsequent Topical Meetings and reports used parameters with fluctuating parameters and values, often without explanation and justification. For example, the October 1999 Topical Meeting used values that were just selected as examples, without any stated basis or criteria for selection. The November 1999 Topical Meeting discussed the ISM Cycle 1 results while the May 2000 Topical Meeting discussed the ISM Cycle 2 results. Both ISM cycles were not based on the design existing at their respective times.

The Firm Fixed Price (FFP) submittal did not contain any hazards or safety analyses because DOE and the Contractor had intended it to be a design basis type of document, suitable for supporting cost and safety analyses but not necessarily including them. However, the NRC staff participated with DOE in the review of the FFP submittal from the perspective of it supporting the Construction Authorization Request/Preliminary Safety Analysis Report (CAR/PSAR). The NRC staff agreed with DOE that the facility and process design information provided in the FFP falls far short of the level of information required to support the CAR/PSAR (see Section 2.5).

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[34] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, March 18, 1999.

3.3.4 NRC ANALYSIS AND EVALUATION

The NRC staff used generic and conceptual process approaches proposed by DOE contractors to analyze potential risks from process and materials aspects at potential TWRS-P facilities (Chapter 7.0, Main References 18 and 19). These analyses identified the following areas of concern:

1. Radiochemical inventories.
2. Process efficacy.
3. Organic ion exchange resin/nitrate interactions.
4. CST drying.
5. Organic materials .
6. Radiolysis.
7. High temperature operations.
8. Nonradioactive chemical effects upon radiochemical processing.

Several of these areas of concern involved events that could be analyzed at this early stage of design using a conservative bounding approach suitable for an initial assessment of risk, determination of relative importance, and the preliminary categorization of SSCs and identification of controls. The U.S. Nuclear Regulatory Commission (NRC) staff used parameters from the privatization contracts (Chapter 7.0, Main References 20 and 21) to estimate inventories and assess materials at risk. NRC and DOE accident handbooks (Main References 22 and 23) outlined several scenarios (e.g., spray leak, tank rupture) with suggested values for release parameters (e.g., atmospheric release fractions, respirable fractions); the staff primarily selected bounding values as these were recommended by the handbooks for preliminary accident analyses on conceptual designs. Dispersion calculations focused on the receptor at 100 meters as this is a typical distance to a member of the public in NRC licensee analyses, and it approximately corresponds to the likely distance from a surface release to the facility fenceline at the proposed, TWRS/Waste Treatment Plant (WTP) location on the Hanford site. The calculations assumed a breathing rate corresponding to light activity. This allowed the determination of dose consequences from these various scenarios.

The NRC staff estimated risk using the LNT (linear no threshold) model for dose consequences, without any modification in the risk factor for acute doses above 10 rem. Frequencies were based upon published values in the literature (see Chapter 7.0, Main References 18 and 19).

Table 4 summarizes potential consequences from unmitigated events, and Table 5 lists potential mitigating controls and their beneficial impacts. All of the accidents listed in Table 4 have potential consequences exceeding the thresholds and guidelines in regulations, including the revised 10 CFR Part 70, referred to as the "New Part 70" in Tables 4 and 5. Many of the events have frequencies in the 1E-2 to 1E-4 range and would be considered to reside in the "unlikely" probability bin.

Using the U.S. national average for workplace fatalities of 4.8E-5/yr (see Main Reference 18) for comparison, ten process scenarios exceed that national average (at 100 meters). The total estimated, unmitigated risk from a generic, TWRS/WTP facility at 100 meters due to these incidents involving radionuclides is approximately 2.4E-2/yr, some 500 times larger than the U.S. workplace average risk. For contrast and comparison, Table 6 displays additional risk comparisons, and shows that the U.S. average background radiation dose dominates individual public radiological risk (at 1.8E-4/yr). Table 6 also lists the average risk due to cancer (2E-3/yr - see Main Reference 18). By comparison, the potential unmitigated risk from these process accidents at a TWRS/WTP facility exceeds the background dose risk by two orders of magnitude and the average cancer risk by a factor of ten. Four accident scenarios involving two forms of melter failure, and two forms of resin interactions dominate the risk by accounting for 90 percent of the total. A large portion of the risk from the two melter accident scenarios accrues from rapid thermal volatilization and dispersal of the aqueous cold cap from a catastrophic release of the high temperature, molten glass. Limited experimental data and experience are available for these melter failure scenarios. If these melter and resin accidents are effectively prevented and/or mitigated, the TWRS-P risk decreases to around 1.4E-3/yr, a level commensurate with the risk associated with occupational exposure limits, but still some 10 times greater than the risk due to average background exposure to radiation. Several accident scenarios involving tank failures or deflagrations also exhibit the potential for very high doses. In the case of chemical storage tank failure, the potential ammonia and nitric acid releases would result in irreversible, deterministic health effects around the TWRS-P facility and its environs, and would render the facility uninhabitable for operating and control purposes. If liquid anhydrous ammonia were used, the affected area could extend out beyond a mile. Thus, prevention and mitigation are required to minimize the impact of these chemical effects upon radioactive materials.

The NRC staff investigated the availability of controls for reducing the risks. Fortunately, relatively simple and effective, prevention and mitigation methods are available, and Table 5 displays this situation. Prevention and mitigation methods reduce the total risk to the receptor at 100 meters from the TWRS-P facility to about 2.5E-6/yr. This result is about 5 percent of the average occupational risk and around 1.4 percent of the risk due to the average background dose. Incorporation of prevention and mitigation controls is likely to be acceptable to the revised Part 70, although further analysis may be necessary for the melter failure scenarios. Consequently, the preventative and mitigating design features are likely to become controls and items relied upon for safety, and potential examples are discussed as follows:

1. Low Activity Waste (LAW) and High Level Waste (HLW) Tank Leak Events: Prevention could be accomplished by high quality tanks and components, with 100 percent weld inspection and leak testing, and inventory controls. The tanks and the ullage ventilation system (including the first high efficiency particulate air (HEPA) filter bank) could become the primary mitigative control. The second ullage ventilation system HEPA filter bank and the cell, with its ventilation system and (single-stage) HEPA filter bank and sump, could constitute likely secondary controls. In order to meet defense in depth requirements, the area outside the cell (usually an access or operating corridor) and its ventilation system, and a short exhaust stack represent probable tertiary controls. Specific level and spill detection instrumentation and alarms may also be identified as controls. Ideally, the inventory of liquid LAW and HLW in the TWRS-P facility should be minimized, perhaps by some form of an inventory limit (e.g., such as a potential limiting conditions of operation (LCO)). A specific design might include several smaller tanks (instead of one large one), of which one or more are maintained empty as a dedicated spare(s) in case of a leak in a full tank. The requirement for a spare tank or spare tank capacity could become a requirement (perhaps another potential LCO).

2. <u>Cesium Tank Events:</u> Prevention could be accomplished by high quality tanks and components, with 100 percent weld inspection and leak testing, and inventory controls. The cesium product tank(s) and the ullage ventilation system (including the first HEPA filter bank) are likely to become the primary control. The second ullage ventilation system HEPA filter bank and the cell, with its ventilation system and (single-stage) HEPA filter bank and sump, constitute probable secondary controls. In order to meet defense in depth requirements, the area outside the cell (usually an access or operating corridor) and its ventilation system, and a short exhaust stack represent likely tertiary controls. If these are (physically) the same items relied upon for safety by the LAW and HLW tanks, then an analysis should be performed demonstrating no deleterious effects upon the safety functions from the different tank systems and their events. Specific level and spill detection instrumentation and alarms could also be identified as controls. Ideally, the inventory of liquid cesium solution in the TWRS-P facility should be minimized or even eliminated, perhaps with an inventory limit (a potential LCO). It is also possible to add a requirement for conversion of the cesium into a solid, such as loading onto CST. A specific design might include several smaller tanks (instead of one large one), of which one or more are maintained empty as a dedicated spare(s) in case of a leak in a full tank. The spare tank or spare capacity could also become a requirement (e.g., another potential LCO).

   The cesium solution in the tank requires cooling to prevent temperature increases and, ultimately, avoid solution boiling. Thus, the cooling means and its backup potentially become items relied upon for safety. This includes the separate cooling coils or jackets (zones), the cooling sources (usually cooling water, with process and firewater backup), the pumps or means of recirculation, the piping, and redundant flow/temperature instrumentation. Interruption of cooling is likely to become an LCO, and the respective temperatures (of the tank and cooling means) Operational Safety Requirements (OSRs). If it could be demonstrated by calculation or experimentation that a tank design is adequately cooled by passive means under all credible conditions, then this would reduce the number of items relied upon for safety.

3. <u>Ion Exchange Column Events:</u> Different controls can be used to provide adequate assurances of safety with the ion exchange columns. Preventative controls rely upon cooling, detection, and dilution/dispersal of the reacting resin mass. Resin performance improves with lower feed solution temperatures (20-25°C or lower); cooler temperatures also reduce the probability of resin degradation reactions. The cooling means, heat exchanger, and temperature indicators are likely to become items relied upon for safety. The column itself is likely to be a preventative control, because it confines a significant amount of radioactivity (i.e., once loaded - to avoid a loss of confinement) and the design may incorporate specific design features to cool or separate the resin bed and prevent runaway resin reactions. If mitigation relies upon an enclosure around the columns, venting to the cell; then the enclosure, vents/relief valves, rupture disks, and the ventilation systems (both the process/vessel ullage [2 HEPA banks in series] and cell [1 HEPA bank]) become the primary controls. Analyses should be performed to verify no cross effects upon other vessels on the same vent header from potential fumes caused by the degradation reactions; if such effects are discovered, then the resin columns should have their own ventilation system. In order to meet defense-in-depth requirements, the area outside the cell (usually an access or operating corridor) and its ventilation system, and a

short exhaust stack represent potential secondary mitigating controls. Specific level and spill detection instrumentation and alarms could also be identified as controls.

4. Tank Deflagration Events: These probably require the same items relied upon for safety as with the tank leak events. Hydrogen and flammable gas detection systems constitute additional potential controls. Redundant exhaust fans, supply air fans, or nitrogen blanket systems may be warranted as controls if more refined calculations or experimental results indicate higher evolution rates for flammable gases.

5. CST Drying Events: Prevention could use OSRs on the CST drying. Temperature and humidity instrumentation are likely to become the primary controls, with temperature and humidity endpoints determined by experimental testing. The CST canister drying enclosure and its filtered vent represent the first potential mitigating control but, in the absence of a specific design, are not credited with any source amelioration. The cell and its ventilation system (including two HEPA filter banks) likely become the secondary control; the first HEPA filter bank is assumed to be rendered ineffective by the deflagration. In order to meet defense-in-depth requirements, the area outside the cell (usually an access or operating corridor) and its ventilation system, and a short exhaust stack represent probable tertiary controls.

6. Melter Events: Prevention could rely upon melter instrumentation and controls to detect maloperation and terminate the radioactive feed to the melter and replace it with process water (to maintain the cold cap), and terminate operations prior to melter release or failure. The proposed design had an inner gap with unspecified instrumentation; melter electrical and thermal sensors would become the probable controls. In the absence of a specific design and details, these are not credited with source prevention and reduction in this generic analysis by the NRC staff. The melter, its enclosure and its filtered vent would likely represent the first potential mitigating control but, in the absence of a specific design, are not credited with any source amelioration. The cell and its ventilation system (including two HEPA filter banks) would probably become the primary control. In order to meet defense-in-depth requirements, the area outside the cell (usually an access or operating corridor) and its ventilation system, the melter offgas system, and a short exhaust stack represent likely secondary controls.

7. Chemical Events: Prevention would use high quality tanks and components with 100 percent weld inspection, and leak testing, and a generous (conservative) corrosion allowance. Prevention also could use an enclosure (with a spill basin) for the reduction of weather effects (including diurnal thermal cycles) upon both the components and the chemicals of concern (principally ammonia and nitric acid). Leak detection sensors and a water spray/deluge system probably constitute the primary mitigative method. Exhaust fans (on the enclosure) activated by a separate sensor system denote a likely secondary mitigative method. The third mitigative method locates the cold chemical storage tank area away from the facility and preferably near a cooling pond or its equivalent. Potential catastrophic releases of chemicals may affect the operability of the TWRS-P facility, rendering the area uninhabitable for a period of time (possibly as long as several hours). Consequently, additional safety requirements may be needed for control room air, breathing air (self contained breathing apparatus), and/or the ability of the TWRS process to operate and shutdown automatically without human assistance.

The average worker and cancer risks presented in Table 6 include contributions from all sources, such as industrial accidents, environmental chemical exposures, and other nonradiological contributors. Therefore, acceptable limits for potential contributions from radiological risks associated with process hazards of TWRS-P are likely to be lower, perhaps a few percent of these averages. This preliminary analysis suggests this is indeed the case for a TWRS-P facility design incorporating standard nuclear industry prevention and mitigation techniques; the estimated risk with prevention and mitigation features is 5 percent of the average occupational risk and 0.1 percent of the average, public cancer risk. This is consistent with discussions in the literature (Reference 18).

From this review of a generic facility, the NRC staff concluded that safety controls (IROFS) would be needed at the proposed TWRS/WTP facility to meet likely risk goals, and that, with the possible exception of the melter areas, no unusual or special controls with unique characteristics would be necessary. More information is needed on the melter designs before specific control strategies can be postulated and evaluated.

In general, designs proposed by the contractors do not consider prevention and controls and only incorporate one mitigating means to overcome failures. The designs do not include important auxiliary effects in the analyses, such as common mode failures, operability, recoverability, and plant habitability for operators, and means for controlling these effects.

Obviously, DOE and its contractors will include experimental testing as part of the program leading to the design, construction, and operation of the TWRS-P facility. Few appropriate safety related parameters, such as failure rates, modes, and release fractions, are available for HLW processing and vitrification facilities. It would be beneficial if the measurement of such safety parameters could be included in the DOE and contractor programs.

3.3.5 PRESENT STATUS OF HAZARDS AND SAFETY ANALYSES

An ISM Cycle 3 was originally planned for late Summer of 2000, based upon the FFP design from the April 26, 2000, submittal. However, this is unlikely to occur due to the contract termination and the selection of new contractors.

It should be noted that many of the original questions and comments raised in the SRD, ISAR, DSF, FFP, and other reviews remain without adequate closure. These include the following:

1. Incomplete process and hazard descriptions and a relative lack of information, details, specificity, and numerical values on many features, including IROFSs.
2. Concerns about adequate margin, conservatism, and defense-in-depth resulting in potential under-estimation of consequences and IROFSs.
3. Design basis not identified vis-a-vis "best basis," average, or median.
4. Uneven approach to dose methodology including the handling of unmitigated events.
5. Inconsistency of unmitigated and mitigated analyses.

Table 4: Summary of Unmitigated Events at the Generic TWRS Facility

Event	Unmitigated Consequence Impact, Receptor at 100 meters, rem	Part 70 Consequence Category	Estimated Frequency (uncontrolled), Event/yr	Likelihood (probability) Bin	Unmitigated Impact, Risk, yr-1
LAW Tank Failure	3,000-6,300	High	2E-5	Unlikely	3E-5 to 6E-5
HLW Tank Failure	6,000-12,000	High	2E-5	Unlikely	6E-5 to 1.2E-4
Cesium Tank - Loss of Cooling/ Boiling (1,000 gal)	25,000	High	1E-6	Highly Unlikely	1.25E-5
Cesium Tank Failure, 1,000 gal Basis	55	High	2E-5	Unlikely	5.5E-7
Cesium Loaded Resin/Nitrate Interaction	1,400	High	1E-3	Unlikely	7E-4 (3% of total, unmitigated risk)
Cesium Eluting, Resin/Nitrate Interaction	3,400	High	1E-3	Unlikely	1.7E-3 (7% of total, unmitigated risk)
H2 Deflag./ LAW Heel	20,000	High	1E-5	Unlikely	1E-4

Table 4: Summary of Unmitigated Events at the Generic TWRS Facility (continued)

Event	Unmitigated Consequence Impact, Receptor at 100 meters, rem	Part 70 Consequence Category	Estimated Frequency (uncontrolled), Event/yr	Likelihood (probability) Bin	Unmitigated Impact, Risk, yr-1
H2 Deflag/ HLW Heel	38,000	High	1E-5	Unlikely	1.9E-4
H2 Deflag/ Cesium Tank Heel	2,500	High	1E-5	Unlikely	1.2E-5
H2/LAW Tank Deflag/low H2	2,300	High	1E-5	Unlikely	1.15E-5
H2/LAW Tank Deflag/high H2	115,000	High	1E-6	Highly Unlikely	6E-5
H2/HLW Tank Deflag/high H2	216,000	High	1E-6	Highly Unlikely	1.1E-4
CST Drying/ H2 Deflag	48,000	High	1E-5	Unlikely	2.4E-4 (1% of total, unmitigated risk)

Table 4: Summary of Unmitigated Events at the Generic TWRS Facility (continued)

Event	Unmitigated Consequence Impact, Receptor at 100 meters, rem	Part 70 Consequence Category	Estimated Frequency (uncontrolled), Event/yr	Likelihood (probability) Bin	Unmitigated Impact, Risk, yr-1
Melter/Canister Failure, Cold Cap Dispersal	14,500	High	1E-3	Unlikely	7.3E-3 (30% of total, unmitigated risk)
Melter/Steam Explosion	26,000	High	1E-3	Unlikely	1.3E-2 (54% of total, unmitigated risk)
Chemical - Ammonia Tank Failure	> ERPG-3	High	1E-5	Unlikely	(not applicable)
Chemical - Nitric Acid Tank Failure	> ERPG-3	High	1E-5	Unlikely	(not applicable)

Total, unmitigated risk = 2.4E-2/yr

Table 5: Summary of the Impact of Potential Controls at the Generic TWRS Facility

Event	Potential Controls/Items Relied Upon For Safety	Potential Mitigated Consequence to Receptor at 100 meters, rem	Mitigated Consequence Category	Mitigated Likelihood (Probability)	Likely Acceptable per New Part 70?	Mitigated Risk, year-1
LAW Tank Failure	1. Tank 2. cell/HEPA 3. Enclosure/sump 4. Spare tank	3-6	Intermediate	2E-6	Yes	3E-9 to 6E-9
HLW Tank Failure	1. Tank 2. cell/HEPA 3. Enclosure/sump 4. Spare tank	6-12	Intermediate	2E-6	Yes	6E-9 to 1.2E-8
Cs Tank LOCA, Boiling/1,000 gal	1. cell/vent./2 HEPA 2. emerg. cooling	25 (first HEPA fails due to moisture)	Intermediate to High	1E-6	Yes	1.3E-8
Cs Tank Failure, 1,000 gal	1. Tank 2. cell/vent. 3. Enclosure/sump 4. Spare tank 5. Cs as solid	0.1	Low	2E-6	Yes	1E-10
Cs Resin, Loaded	1. Enclosure 2. Cell/vent/HEPA	1.4	Low	1E-4	Yes	7E-8
Cs Resin, Elution	1. Enclosure 2. Cell/vent/HEPA	3.4	Low	1E-4	Yes	1.7E-7 (7% of total)
H2 Deflag/ LAW Tank Heel	1. Gas/vent./2 HEPA 2. Cell/vent 3. Sensor/N2 inject	20 (first HEPA rendered ineffective)	Intermediate	1E-6	Yes	1E-8

Table 5: Summary of the Impact of Potential Controls at the Generic TWRS Facility (continued)

Event	Potential Controls/Items Relied Upon For Safety	Potential Mitigated Consequence to Receptor at 100 meters, rem	Mitigated Consequence Category	Mitigated Likelihood (Probability)	Likely Acceptable per New Part 70?	Mitigated Risk, year-1
H2 Deflag/ HLW Tank Heel	1. Gas/vent./2 HEPA 2. Cell/vent 3. Sensor/N2 inject	38 (first HEPA rendered ineffective)	Intermediate	1E-6	Yes	2E-8
H2 Deflag/ Cs Tank Heel	1. Gas/vent./2 HEPA 2. Cell/vent 3. Sensor/N2 inject	2.5 (first HEPA rendered ineffective)	Low	1E-6	Yes	1.3E-9
H2 LAW Tank Deflag/ Low H2	1. Gas/vent./2 HEPA 2. Cell/vent 3. Sensor/N2 inject	2.3 (first HEPA rendered ineffective)	Low	1E-6	Yes	1.2E-9
H2/LAW Tank Deflag/ High H2	1. Gas/vent./2 HEPA 2. Cell/vent 3. Sensor/N2 inject	115 (first HEPA rendered ineffective)	High	1E-6	Yes	6E-8
H2/HLW Tank Deflag/ High H2	1. Gas/vent./2 HEPA 2. Cell/vent 3. Sensor/N2 inject	216 (first HEPA rendered ineffective)	High	1E-6	Yes	1.1E-7 (4.3% of total)

Table 5: Summary of the Impact of Potential Controls at the Generic TWRS Facility (continued)

Event	Potential Controls/Items Relied Upon For Safety	Potential Mitigated Consequence to Receptor at 100 meters, rem	Mitigated Consequence Category	Mitigated Likelihood (Probability)	Likely Acceptable per New Part 70?	Mitigated Risk, year-1
CST Drying/ H2 Deflag	1. Cell/vent/2 HEPA 2. Enclosure	48 (first HEPA rendered ineffective)	High	1E-6	Yes	2.4E-8
Melter/Canister/ Cap Dispersal	1. Cell/vent/2 HEPA 2. Instrumentation	15 (first HEPA rendered ineffective due to heat)	Intermediate	1E-4	Further Analysis Necessary	7.5E-7 (30% of Total)
Melter/Steam Explosion	1. Cell/vent/2 HEPA 2. Instrumentation	26 (first HEPA rendered ineffective due to blast and heat)	Intermediate to High	1E-4	Further Analysis Necessary	1.3E-6 (51% of Total)
Ammonia Tank Failure	1. Tank 2. Enclosure 3. Detect/Sprays	< ERPG-1	Low	1E-6	Yes	(0)
Nitric Acid Tank Failure	1. Tank 2. Enclosure 3. Detect/Sprays	< ERPG-1	Low	1E-6	Yes	(0)

Total mitigated risk = 2.5E-6/yr

Table 6: Different Sources of Risk Limits

Risk Source/Basis	Dose Equivalent, Rem	Frequency, yr-1	Risk, yr-1
Worker Limits			
Part 20, Worker Limit	5	1	2E-3
Part 20, Typical ALARA Value	0.31	1	1.2E-4
U.S. Worker Average, All Causes	(-)	(-)	4.8E-5
Public Limits			
Part 20, Public Limit	0.1	1	5E-5
Part 20, D&D and Part 61, Public Limits	0.025	1	1.3E-5
Typical Public Values			
U.S. Average Background	0.350	1	1.8E-4
Background Difference between Denver and U.S. Average	0.500	1	2.5E-4
Average U.S. Public Cancer Fatality Rate	(NA)	(NA)	2E-3
Average Public Dose from Commercial Nuclear Plant	<0.001	1	< 5E-7

Note: Radiological comparisons assume Linear No Threshold (LNT) theory, with risk factors of 2,500 rem/fatality for workers and 2,000 rem/fatality for members of the public. These rates are kept constant, and not reduced for higher acute doses (e.g., 1,000 rem/fatality for individual, acute doses over 10 rem).

6. Use of optimistic design assumptions and reliabilities.

7. Uncertainties and inconsistencies not addressed.

8. Selection, validation and justification of data, assumption, and models.

9. Emphasis on active instead of passive mitigation and control.

10. Organization of documents and information.

11. Inconsistencies in hazards identification, SSC categorization, and design classes

12. Relative lack of quantification.

13. Presence of large tanks and inventories.

14. Limited consideration of offnormal and unanticipated events, and interactions.

In addition, the ISM process itself is used as a risk-based approach, without consideration of a basal set of codes, standards, minimum performance requirements, or basic design criteria. Numerical results are being used in absolute terms for comparisons with dose and frequency limits. No consideration of uncertainties by either analysis or statistics (e.g., 90th percentile) has been included in the work to date - the emphasis has been on expected, "best basis," or median (50th percentile). This expresses greater credibility in the capability of current modeling techniques, the quality of the available parameters, and the level of detail available in the design (currently at a circa 15 percent level) than appears warranted at this time. As a result, it raises doubts that defense-in-depth is adequately followed and that the IROFSs are fully identified with this approach. The ISM approach as followed at TWRS-P does not appear to be consistent with the NRC's risk-informed, performance-based approach.

NRC analyses indicate significant radioactive and chemical inventories at the proposed facility that translate into potentially high levels of risk unless prevention and mitigation methods are used to address accident scenarios. This implies that a significant number of IROFSs will be needed. Potential IROFSs include those typical for fuel cycle facilities such as HEPA filter banks. NRC analyses also show a significant distance effect. Thus, analyses conducted using NRC-like approaches of 100 meters or so to a public receptor (the facility fence line) result in higher estimated consequences and more IROFSs as compared to the DOE approach of using the Hanford site's security perimeter. The latter corresponds to distances of 11,000 to 15,000 meters and reduces the consequences by 100-1000 (due to dilution), resulting in fewer IROFSs. Given that personnel are currently onsite that would be considered as members of the public under NRC guidance and that future site conditions will shrink the security perimeter, a distance much shorter than the 11,000-15,000 range would seem to be needed for estimating potential consequences to the public.

Furthermore, changes in the contracting approach may significantly affect the design and hazard analyses. There will now be a minimum of three contractors or contractor teams involved (interim, design and construction, and operations). Thus, it is not clear how the expertise on design, technology, and safety will be transferred from one contractor organization to the other. The new contractors also may not have to follow the existing design and approach, and therefore, some or all of the preceding hazards analyses may no longer be relevant. The proposed new contract will incorporate significant incentive clauses for capital cost reductions. It remains to be seen how these will be balanced vis-a-vis operating requirements and safety issues.

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3.0 NRC AREAS OF REVIEW

3.4 RADIATION SAFETY AND DOSE ASSESSMENT METHODOLOGY

3.4.1 REGULATORY LIMITS FOR OCCUPATIONAL AND PUBLIC DOSES APPLICABLE TO TANK WASTE REMEDIATION SYSTEM-PRIVATIZATION

This chapter presents a comparison of regulatory approach, dose limits, and monitoring thresholds stated in 10 CFR Part 20, "Standards for Protection Against Radiation," and 10 CFR Part 835, "Occupational Radiation Protection."

3.4.1.1 INTRODUCTION

The purpose of this chapter is to discuss the similarities and differences between 10 CFR Part 20, "Standards for Protection Against Radiation" (which affects U.S. Nuclear Regulatory Commission (NRC) licensees), and 10 CFR Part 835, "Occupational Radiation Protection," (which affects U.S. Department of Energy (DOE) facilities), and provide a basis for future activities should a transition to regulatory oversight by the NRC take place at some future time. Such a transition could affect certain aspects of the contractor's Radiation Control Program. The chapter also explains some of the regulatory philosophy of the DOE Regulatory Unit (RU) that is responsible for oversight of the TWRS-P contractor and how this philosophy may differ from that of the NRC.

It should be understood at the outset that the NRC and DOE numerical dose limits are identical for individuals who receive occupational dose, and another set of identical limits applies to members of the public. However, differences exist between 10 CFR Part 20 and 10 CFR Part 835 with respect to definitions of the conditions under which monitoring is required and how persons and situations are classified in terms of the type of monitoring performed.

Misunderstandings may sometimes arise through the usage of terms that are not defined in the regulations, but which are similar to, or have the appearance of formal definitions. These include worker, radiation worker, radiological worker (defined in Part 835 but not in Part 20), and occupationally exposed, among others. Tables are provided that list and compare the definitions contained in the two Parts. A discussion is undertaken of Occupational Dose Limits and Limits for Members of the Public as stated in Parts 20 and 835; it will be seen that the limits are nearly identical. A comparison of Monitoring Thresholds mandated by Parts 20 and 835 is also presented.

The Radiological Exposure Standards Above Normal Background for the TWRS-P Project are derived from 10 CFR Part 835 and the DOE/Regulatory Unit (RU) Top Level Standards, and for this reason do not have a directly corresponding regulation in Part 20 with which to compare. These standards are discussed in Section 3.4.1.3.

For the TWRS-P facility, in addition to the categories of general employee and radiological worker (defined in Part 835) and member of the public (defined in both Parts), a third category known as the co-located worker (CLW) has been defined within the "Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors," DOE/RL-96-0006, Revision 1, July 1998. The CLW is defined in Section 3.4.1.6, Glossary, 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.

3.4.1.2 DEFINITIONS

3.4.1.2.1 MEMBERS OF THE PUBLIC

Definitions related to members of the public are listed in alphabetical sequence in Table 7. From the NRC (10 CFR Part 20) perspective, an individual who is outside a licensed facility's restricted area would normally be considered a member of the public.[35] In contrast, an individual outside a controlled area as defined in 10 CFR Part 835 may be either a member of the public or a CLW, although the concept of CLW is not defined for all DOE facilities, and as stated above, is not contained in 10 CFR Part 835. If such an individual is employed in a neighboring facility where he or she is subject to receiving an occupational dose, that individual may be considered to be a CLW relative to the Contractor's facility, according to DOE. Otherwise, that individual may fall into one of three other categories discussed below under "Individuals Subject to Occupational Dose."

Although both 10 CFR Part 20 and 10 CFR Part 835 contain regulations governing dose limits to members of the public, and although the definitions of "member of the public" are similar in both, the regulatory language differs in several respects. Part 20 of 10 CFR provides limits for members of the public in general, while 10 CFR Part 835 provides limits only for those members of the public entering a controlled area. Note that "controlled area" is defined in 10 CFR Part 835 differently from 10 CFR Part 20, and that the definition of a "restricted area" in 10 CFR Part 20 is very close to that of a "controlled area" in 10 CFR Part 835, although the 10 CFR Part 20 definition of a "restricted area" has no direct parallel in 10 CFR Part 835. The dose limits associated with these definitions are provided in Section 3.4.1.3.

3.4.1.2.2 INDIVIDUALS SUBJECT TO OCCUPATIONAL DOSE LIMITS

While the definitions of "member of the public" are similar in both 10 CFR Part 20 and 10 CFR Part 835, a comparison of the regulations pertaining to occupational dose is not as clear. The definitions in each Part differ with respect to individuals employed as workers in a facility that utilizes radioactive material. This section discusses the contrasting definitions and regulatory approaches of 10 CFR Parts 20 and 835 with respect to occupational dose. Table 8 lists the definitions in alphabetical order and clearly shows where a definition in 10 CFR Part 20 has no equivalent in 10 CFR Part 835, and vice-versa.

Because terms related to "radiation area" are contained in the definitions of Table 8, Section 3.4.1.2.3 provides definitions of such terms in Table 9.

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[35] An exception would be if an individual normally permitted to receive an occupational dose utilizes radioactive materials from the licensed facility outside its restricted area, but under the control of the licensee.

Table 7: Definitions Related to "Member of the Public"

TERM	Part 20 Definitions	Part 835 Definitions
Controlled Area	Controlled area means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.	Controlled area means any area to which access is managed by or for DOE to protect individuals from exposure to radiation and/or radioactive material.
Member of the Public	Member of the public means any individual except when that individual is receiving an occupational dose.	Member of the public means an individual who is not a general employee. An individual is not a "member of the public" during any period in which the individual receives an occupational dose.
Public Dose	Public dose means the dose received by a member of the public from exposure to radiation or radioactive material released by a licensee, or to any other source of radiation under the control of a licensee. Public dose does not include occupational dose or doses received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with §35.75, or from voluntary participation in medical research programs.	*(no corresponding definition)*
Restricted Area	Restricted area means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.	*(no corresponding definition)*
Site Boundary	Site boundary means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.	*(no corresponding definition)*
Unrestricted Area	Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.	*(no corresponding definition)*

To summarize, the 10 CFR Part 20 regulations distinguish only between a "member of the public" (discussed above) and an individual who receives an "occupational dose." An individual may receive an occupational dose from radioactive materials present in the facility where the work is performed without being directly employed by that facility.

In contrast, under the regulations in 10 CFR Part 835, an individual may be a "member of the public," an individual receiving an "occupational dose," a "general employee," or a "radiological worker." An individual need not be a general employee or a radiological worker in order to receive an occupational dose. As with the 10 CFR Part 20 regulations, an individual may receive an occupational dose from radioactive materials present in the facility where the work is performed without being directly employed by that facility. A person may receive an occupational dose even though he or she is not a radiological worker.

The occupational dose limits are the same in both Parts. However, for the TWRS-P facility, a set of dose limits has been established for a series of "event probability ranges," as described in RL/REG-98-18, Revision 0, "Regulatory Unit Position on Radiological Safety for Hanford Co-

Located Workers," September 16, 1998, Exhibit 1: Table 2-1, "Radiological Exposure Standards Above Normal Background." Numerical limits are discussed in Section 3.4.1.3.

In accordance with DOE/RL-96-0006, individuals designated as CLWs are permitted to accrue occupational doses that are greater than that permitted for members of the public.

3.4.1.2.3 RADIATION AREA, HIGH RADIATION AREA, VERY HIGH RADIATION AREA

The previous section provided definitions of terms related to "occupational dose." Some of the terms contained in the definitions in Table 8 require explanation. Table 9 in this section provides definitions of terms related to "radiation area" as defined in Table 8.

3.4.1.2.4 CONTAMINATION AND RADIOACTIVE MATERIAL

Table 10 in this section captures additional definitions relevant to radiological protection contained in 10 CFR Part 20 and 10 CFR Part 835 regulations. These terms are listed because they are related to the means by which occupational radiation doses might be encountered in facilities utilizing radioactive materials. It also serves to point out some of the other differences in the two sets of regulations. Both Parts provide definitions of "Airborne Radioactivity Area," but the remaining terms are defined only in Part 835.

3.4.1.2.5 DEFINITIONS OF TERMS RELATED TO DOSE AND DOSE EQUIVALENT

The definitions contained in this section are provided to remind the reader of the technical terminology that describes how dose is measured. Specific terms have been assigned to differentiate the type of measurement to be quantified. A thorough understanding of these terms is not required in order to appreciate the differences in 10 CFR Part 20 versus 10 CFR Part 835 radiological protection standards. In Table 11, the definitions are given in a hierarchical format with each definition building on its antecedent. Note that in 10 CFR Parts 20 and 835 there are small differences in the way these definitions are phrased, but that they are equivalent in all other respects.

3.4.1.3 LIMITS

Table 12 provides a comparison of the limits set by 10 CFR Part 20 and 10 CFR Part 835. Note that the two sets of limits are effectively the same.

As indicated in Table 12, the annual dose limit for members of the public is the same in both Parts, namely 0.1 rem (0.001 sievert). As stated previously, 10 CFR Part 835 is concerned only with those members of the public who enter a controlled area, while Part 20 covers any member of the public. Part 20 of 10 CFR also provides a limit of 0.002 rem (0.02 millisievert) in any one hour for the dose in an unrestricted area. There is no corresponding definition in 10 CFR Part 835. Note that while 10 CFR Part 835 is concerned with members of the public who enter a <u>controlled</u> area (as defined in 10 CFR Part 835), the situation is not analogous to 10 CFR Part 20, since 10 CFR Part 20 concerns members of the public outside any <u>restricted</u> area (as defined in 10 CFR Part 20).

Table 8: Definitions Related to "Occupational Dose"

TERM	Part 20 Definitions	Part 835 Definitions
Controlled Area	Controlled area means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.	Controlled area means any area to which access is managed by or for DOE to protect individuals from exposure to radiation and/or radioactive material.
General Employee	*(no corresponding definition)*	General employee means an individual who is either a DOE or DOE contractor employee; an employee of a subcontractor to a DOE contractor; or an individual who performs work for or in conjunction with DOE or utilizes DOE facilities.
Occupational Dose	Occupational dose means the dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include dose received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with §35.75, from voluntary participation in medical research programs, or as a member of the public.	Occupational dose means an individual's ionizing radiation dose (external and internal) as a result of that individual's work assignment. Occupational dose does not include doses received as a medical patient or doses resulting from background radiation or participation as a subject in medical research programs.
Radiological Area	*(no corresponding definition)*	Radiological area means any area within a controlled area defined in this section as a "radiation area," "high radiation area," "very high radiation area," "contamination area," "high contamination area," or "airborne radioactivity area."
Radiological Worker	*(no corresponding definition)*	Radiological worker means a general employee whose job assignment involves operation of radiation producing devices or working with radioactive materials, or who is likely to be routinely occupationally exposed above 0.1 rem (0.001 sievert) per year total effective dose equivalent.
Restricted Area	Restricted area means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.	*(no corresponding definition)*
Site Boundary	Site boundary means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.	*(no corresponding definition)*
Unrestricted Area	Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.	*(no corresponding definition)*

Table 9: Definitions Related to "Radiation Area"

TERM	Part 20 Definitions	Part 835 Definitions
Radiation Area	Radiation area means an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.	Radiation area means any area accessible to individuals in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.005 rem (0.05 millisievert) in 1 hour at 30 centimeters from the source or from any surface that the radiation penetrates.
High Radiation Area	High radiation area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.	High radiation area means any area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.1 rem (0.001 sievert) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.
Very High Radiation Area	Very high radiation area means an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in 1 hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates. (Note: At very high doses received at high dose rates, units of absorbed dose (e.g., rads and grays) are appropriate, rather than units of dose equivalent (e.g., rems and sieverts)).	Very high radiation area means any area accessible to individuals in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in one hour at 1 meter from a radiation source or from any surface that the radiation penetrates.

Table 10: Definitions Related to "Contamination" and "Radioactive Material"

TERM	Part 20 Definitions	Part 835 Definitions
Airborne Radioactivity Area	Airborne radioactivity area means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations -- (1) In excess of the derived air concentrations (DACs) specified in Appendix B, to §§20.1001 - 20.2401, or (2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake (ALI) or 12 DAC-hours.	Airborne radioactivity area means any area, accessible to individuals, where: (1) The concentration of airborne radioactivity, above natural background, exceeds or is likely to exceed the derived air concentration (DAC) values listed in Appendix A or Appendix C of this part; or (2) An individual present in the area without respiratory protection could receive an intake exceeding 12 DAC-hours in a week.
Contamination Area	*(no corresponding definition)*	Contamination area means any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed the removable surface contamination values specified in Appendix D of this part, but do not exceed 100 times those values.
High Contamination Area	*(no corresponding definition)*	High contamination area means any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed 100 times the removable surface contamination values specified in Appendix D of this part.
Radioactive Material Area	*(no corresponding definition)*	Radioactive material area means any area within a controlled area, accessible to individuals, in which items or containers of radioactive material exist and the total activity of radioactive material exceeds the applicable values provided in Appendix E of this part.

Table 11: Definitions Related to "Dose"

TERM	Part 20 Definition	Part 835 Definitions
Dose	Dose or radiation dose is a generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent, as defined in other paragraphs of this section.	Dose is a general term for absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent as defined in this part.
Dose Equivalent H_T	Dose equivalent (H_T) means the product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The units of dose equivalent are the rem and sievert (Sv).	Dose equivalent (H) means the product of absorbed dose (D) in rad (or gray) in tissue, a quality factor (Q), and other modifying factors (N). Dose equivalent is expressed in units of rem (or sievert) (1 rem = 0.01 sievert).
Effective Dose Equivalent H_E	Effective dose equivalent (H_E) is the sum of the products of the dose equivalent to the organ or tissue (H_T) and the weighting factors (w_T) applicable to each of the body organs or tissues that are irradiated ($H_E = \Sigma w_T H_T$).	Effective dose equivalent (H_E) means the summation of the products of the dose equivalent received by specified tissues of the body (H_T) and the appropriate weighting factor (w_T)--that is, $H_E = \Sigma w_T H_T$. It includes the dose from radiation sources internal and/or external to the body. For purposes of compliance with this part, deep dose equivalent to the whole body may be used as effective dose equivalent for external exposures. The effective dose equivalent is expressed in units of rem (or sievert).
Committed Dose Equivalent $H_{T,50}$	Committed dose equivalent ($H_{T,50}$) means the dose equivalent to organs or tissues of reference (T) that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.	Committed dose equivalent ($H_{T,50}$) means the dose equivalent calculated to be received by a tissue or organ over a 50-year period after the intake of a radionuclide into the body. It does not include contributions from radiation sources external to the body. Committed dose equivalent is expressed in units of rem (or sievert).
Committed Effective Dose Equivalent $H_{E,50}$	Committed effective dose equivalent ($H_{E,50}$) is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.	Committed effective dose equivalent ($H_{E,50}$) means the sum of the committed dose equivalents to various tissues in the body ($H_{T,50}$), each multiplied by the appropriate weighting factor (w_T)--that is, $H_{E,50} = \Sigma w_T H_{T,50}$. Committed effective dose equivalent is expressed in units of rem (or sievert).
Total Effective Dose Equivalent TEDE	Total Effective Dose Equivalent (TEDE) means the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).	Total effective dose equivalent (TEDE) means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Table 12: Occupational Dose Limits and Limits for Members of the Public

Category	Part 20 Dose Limits	Part 835 Dose Limits
Occupational (Total Effective Dose Equivalent and Organ or Tissue)	(1) An annual limit, which is the more limiting of: (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).	(1) A total effective dose equivalent of 5 rems (0.05 sievert); (2) The sum of the deep dose equivalent for external exposures and the committed dose equivalent to any organ or tissue other than the lens of the eye of 50 rems (0.5 sievert);
Occupational (Lens of Eye, Skin, Extremities)	(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are: (i) A lens dose equivalent of 15 rems (0.15 Sv), and (ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.	(3) A lens of the eye dose equivalent of 15 rems (0.15 sievert); and (4) A shallow dose equivalent of 50 rems (0.5 sievert) to the skin or to any extremity.
Public	0.1 rem (1 millisievert) 0.002 rem (0.02 millisievert) in any one hour for the dose in an unrestricted area	0.1 rem (0.001 sievert) *(no corresponding definition)*

A more detailed explanation of the dose limits applied to members of the public and to individuals subject to occupational dose, which includes CLWs, needs to include a review of the four event probability ranges, as described in RL/REG-98-18, Revision 0, "Regulatory Unit Position on Radiological Safety for Hanford Co-Located Workers." These are reproduced here as Table 13, Radiological Exposure Standards Above Normal Background.

Table 13 specifies four event probability ranges addressing normal operation and credible accident conditions. General guidelines and frequencies listed for the four event probability ranges are:

1. Normal events are typical of normal facility operations expected to occur regularly in the course of facility operations; the associated frequency of occurrence during the lifetime of the facility is 1 or more per year. As defined in Table 13, a general guideline for this event probability is that normal modes of operating the facility systems should provide adequate protection of health and safety.

2. Anticipated events are characterized as minor incidents and upsets of moderate frequency that may occur once or more during the lifetime of the facility; the associated probability range is $1x10^{-2}$ to <1 per year. As defined in Table 13, a general guideline for this event probability range is that the facility should be capable of returning to operation without extensive corrective action or repair.

3. Unlikely events are characterized as more severe incidents that are not expected, but may occur, during the lifetime of the facility; the associated probability range is $1x10^{-4}$ to $1x10^{-2}$

per year. As defined in Table 13, a general guideline for this event probability range is that the facility should be capable of returning to operation following potentially extensive corrective action or repair, as necessary.

4. Extremely unlikely events are characterized as 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; the associated probability range is 1×10^{-6} to 1×10^{-4} per year. As defined in Table 13, a general guideline for this event probability range is that facility damage may preclude returning to operation.

Note that a probability of occurrence of 1×10^{-2} per year is equivalent to a frequency of one occurrence in 100 years; 1×10^{-4} per year equates to one in 10,000 years; and 1×10^{-6} per year equates to one in 1,000,000 years. These time periods are also known as recurrence intervals.

The additional category of worker has implications with regard to the design and construction of structures, systems, and components (SSCs) of the TWRS-P facility, but does not establish a different set of radiation exposure standards (limits) or a different threshold for monitoring these workers. As may be observed by review of Table 13, for all categories of events, the limits are the same for facility workers and for co-located workers (with the exception that facility workers are subject to the additional limits of ≤50 rem/yr to any organ, the skin, or extremity, and ≤15 rem/yr to the lens of the eye).

The benefit of using the co-located worker category is that it allows the TWRS-P facility to treat these workers as individuals subject to occupational dose, rather than as members of the public under accident conditions. Therefore, in the design and construction of TWRS-P facility SSCs, the "offsite" consequences (i.e., the radiation doses to individuals beyond the boundaries of the TWRS-P facility) are permitted to be greater for the nearby co-located workers than would be the case if these personnel were considered to be members of the public

For the category of normal events, which have a frequency of occurrence of one or more per year, and are considered to have the lowest consequence in terms of harm, co-located workers will seldom, if ever, receive any additional radiation exposure from the TWRS-P facility. However, operation under non-normal conditions, i.e., with releases above ALARA (as low as reasonably achievable) levels, may add to the exposure received by co-located workers. It should be noted that co-located workers will continue to be subject to the same limits as the worker, so that any exposure received due to operation of TWRS-P (under non-normal conditions) is added to that received from the duties performed in the facility in which their occupational exposure is normally received.

DOE regulations pertaining to occupational radiation protection (Part 835) provide some specificity beyond that found in NRC regulations (Part 20), due to the inclusion of certain definitions in Part 835, including those for "general employee" and "radiological worker."

Table 13: Radiological Exposure Standards Above Normal Background

Description	Estimated Frequency of Occurrence f (yr⁻¹)	General Guidelines	Worker	Co-Located Worker	Public
Normal Events: Events that occur regularly in the course of facility operation (e.g., normal facility operations); including routine and preventive maintenance activities.	>0.1	Normal modes of operating facility systems should provide adequate protection of health and safety.	≤ 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)	≤ 10 mrem/yr (airborne pathway) ≤ 100 mrem/yr (all sources) ≤ 100 mrem/yr (public in the controlled area) ≤ 25 mrem/yr (radioactive waste)
Anticipated Events: Events of moderate frequency that may occur once or more during the life of a facility (e.g., minor incidents and upsets).	$10^{-2} < f \leq 10^{-1}$	The facility should be capable of returning to operation without extensive corrective action or repair.	≤ 5 rem/event (2,3) 1.0 rem/event design action threshold (4)	≤ 5 rem/event (2, 3) 1.0 rem/event design action threshold (4)	≤ 100 mrem/event (3)
Unlikely Events: Events that are not expected, but may occur during the lifetime of a facility (e.g., more severe incidents).	$10^{-4} < f \leq 10^{-2}$	The facility should be capable of returning to operation following potentially extensive corrective action or repair, as necessary.	≤ 25 rem/ event (2,3)	≤ 25 rem/event (2, 3)	≤ 5 rem/event (3)
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	$10^{-6} < f \leq 10^{-4}$	Facility damage may preclude returning to operation.	≤ 25 rem/ event (2,3)	≤ 25 rem/event (2,3)	≤ 25 rem/event ≤ 5 rem/event target (3) ≤ 300 rem/event to thyroid
Location of Receptor			Within the BNFL TWRS-P Controlled Area Boundary, including 241-AP-106	The most limiting location at or beyond the BNFL TWRS-P Controlled Area Boundary	The most limiting location along the near river bank/Hwy 240 /southern boundary

(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 worker and co-located worker exposure standards for accidents, the Worker Accident Risk Goal is satisfied through the calculation of the risk from accidents with accident prevention and mitigation features added as necessary to meet the Goal.

(3) In addition to meeting the listed exposure standards for accidents, BNFL Inc.'s 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.

(4) 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 preventive or mitigative features if they are not both cost-effective and feasible.

The importance of these additional definitions in the regulation of the TWRS-P facility is explained in RL/REG-98-18, Revision 0, "Regulatory Unit Position on Radiological Safety for Hanford Co-Located Workers." This document points out that a person can be occupationally exposed even though he or she is not a radiological worker.

A document which explains the use of the categories of "Facility Worker," "Co-Located Worker," and "Public" is BNFL-5193-RES-01, Rev. 0, August 28, 1997, "Radiological and Nuclear Exposure Standards for Facility and Co-Located Workers." These terms are defined in Section 3.4.1.3, "Development of the BNFL Approach to Compliance with Table 1 of DOE/RL-96-0006."

3.4.1.4 MONITORING THRESHOLDS

Generally speaking, the NRC requires monitoring of adults likely to receive in excess of 10 percent of the occupational dose limits, while the DOE requires monitoring of adults likely to receive in excess of 2 percent of the occupational dose limits. For whole body exposure, these equate to 500 mrem and 100 mrem, respectively.

Table 14, "Monitoring Thresholds," provides information from 10 CFR Part 20 and 10 CFR Part 835 relevant to the monitoring of individuals subject to occupational dose and members of the public. These Parts also contain information about monitoring exposure to minors and declared pregnant women (in 10 CFR Part 20) or declared pregnant workers (in 10 CFR Part 835).

3.4.2 RADIATION PROTECTION PROGRAM

This section presents a comparison of regulatory requirements and guidance of the NRC and the DOE with regard to the criteria for Radiation Protection Programs as described in 10 CFR Part 20, Standards for Protection Against Radiation and 10 CFR Part 835, Occupational Radiation Protection.

3.4.2.1 INTRODUCTION

Should a transition to regulatory oversight by the NRC take place at some future time, certain aspects of the contractor's Radiation Control Program could be affected. The objectives of this section are to:

1. Describe the contractor's Radiation Protection Program (RPP) and the regulatory basis for requiring its use.

2. Provide an explanation of differences in the regulatory approach of the two agencies with respect to this programmatic requirement.

3. Identify, insofar as possible, the extent to which the Contractor's RPP would be affected by a transition to NRC regulatory authority.

Table 14: Monitoring Thresholds

Section in 10 CFR:	Part 20	Part 835
	10 CFR 20.1502 Conditions requiring individual monitoring of external and internal occupational dose.	**10 CFR 835.402 Individual monitoring.**
	Each licensee shall monitor exposures to radiation and radioactive material at levels sufficient to demonstrate compliance with the occupational dose limits of this part. As a minimum --	*(no corresponding text)*
External	(a) Each licensee shall monitor occupational exposure to radiation from licensed and unlicensed radiation sources under the control of the licensee and shall supply and require the use of individual monitoring devices by-- (1) Adults likely to receive, in 1 year from sources external to the body, a dose in excess of 10 percent of the limits in §20.1201(a), (2) Minors likely to receive, in 1 year, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem (1 mSv), a lens dose equivalent in excess of 0.15 rem (1.5 mSv), or a shallow dose equivalent to the skin or to the extremities in excess of 0.5 rem (5 mSv); (3) Declared pregnant women likely to receive during the entire pregnancy, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem (1 mSv);[2] and (4) Individuals entering a high or very high radiation area. [2] All of the occupational doses in §20.1201 continue to be applicable to the declared pregnant worker as long as the embryo/fetus dose limit is not	(a) For the purpose of monitoring individual exposures to *external* radiation, personnel dosimeters shall be provided to and used by: (1) Radiological workers who, under typical conditions, are likely to receive one or more of the following: (i) An effective dose equivalent to the whole body of 0.1 rem (0.001 sievert) or more in a year; (ii) A shallow dose equivalent to the skin or to any extremity of 5 rems (0.05 sievert) or more in a year; (iii) A lens of the eye dose equivalent of 1.5 rems (0.015 sievert) or more in a year; (2) Declared pregnant workers who are likely to receive from external sources a dose equivalent to the embryo/fetus in excess of 10 percent of the limit at Sec. 835.206(a); (3) Occupationally exposed minors likely to receive a dose in excess of 50 percent of the applicable limits at Sec. 835.207 in a year from external sources; (4) Members of the public entering a controlled area likely to receive a dose in excess of 50 percent of the limit at Sec. 835.208 in a year from external sources; and (5) Individuals entering a high or very high radiation area.

Section in 10 CFR:	Part 20	Part 835
	exceeded.	
Internal	(b) Each licensee shall monitor the occupational intake of radioactive material by and assess the committed effective dose equivalent to -- (1) Adults likely to receive, in 1 year, an intake in excess of 10 percent of the applicable ALI(s) in Table 1, Columns 1 and 2, of Appendix B to §§20.1001-20.2400; **[see note in last row of this table]** (2) Minors likely to receive, in 1 year, a committed effective dose equivalent in excess of 0.1 rem (1 mSv); and (3) Declared pregnant women likely to receive, during the entire pregnancy, a committed effective dose equivalent in excess of 0.1 rem (1 mSv). *(no corresponding text)*	(c) For the purpose of monitoring individual exposures to *internal* radiation, internal dosimetry programs (including routine bioassay programs) shall be conducted for: (1) Radiological workers who, under typical conditions, are likely to receive a committed effective dose equivalent of 0.1 rem (0.001 sievert) or more from all occupational radionuclide intakes in a year; (2) Declared pregnant workers likely to receive an intake or intakes resulting in a dose equivalent to the embryo/fetus in excess of 10 percent of the limit stated at Sec. 835.206(a); (3) Occupationally exposed minors who are likely to receive a dose in excess of 50 percent of the applicable limit stated at Sec. 835.207 from all radionuclide intakes in a year; or (4) Members of the public entering a controlled area likely to receive a dose in excess of 50 percent of the limit stated at Sec. 835.208 from all radionuclide intakes in a year.
	[Note: 1 ALI = 5 rem, or 5000 mrem for whole body exposure; therefore 10% of the limit (1 ALI) = 500 mrem; also 1 ALI = 50 rem, or 50 000 mrem for the maximally exposed organ; therefore 10% of the limit (1 ALI) = 5 rem = 5000 mrem]	

Table 14: Monitoring Thresholds
(Continued)

The acronym RPP is sometimes used to refer to both "River Protection Project" and to "Radiation Protection Program;" however, in this section, it will be used only in the latter sense, i.e., to refer to the Radiation Protection Program.

3.4.2.2 DESCRIPTION OF THE RADIATION PROTECTION PROGRAM

A document prepared by the former contractor (BNFL Inc.) entitled "Radiation Protection Program for Design" (BNFL-TWP-SER-003) is the radiation program document for achieving compliance with the Code of Federal Regulations (CFR) Part 835, "Occupational Radiation

Protection." As of November 15, 1999, BNFL Inc. had submitted the third revision to this document to the DOE-RU. In a letter from BNFL Inc. to the RU dated January 31, 2000, BNFL Inc. indicated its intention to submit Revision 4 of this document, to be known as the RPP for Design and Limited Construction, on or before May 1, 2000. This was subsequently delayed to June 30, 2000. Upon submittal, it was subjected to an acceptance review by the RU, which rejected it for a number of inadequacies.

The Introduction to BNFL-TWP-SER-003 provides the following description of the RPP:

> The RPP is developed and submitted for regulatory approval in stages corresponding to the status of the [TWRS-P] Project. This initial RPP submittal describes the plans and measures for achieving compliance with the requirements of 10 CFR [Part] 835 that are applicable to the [TWRS-P] Facility design phase. No radiological source term will exist to cause personnel exposures during the design phase; however, decisions made during the design phase will affect exposures during facility operations and deactivation.

3.4.2.3 REGULATORY BASIS

In contrast to 10 CFR Part 20, the regulations in 10 CFR 835.101 are very prescriptive with respect to the requirement for a radiation protection program. While both agencies require the implementation of an RPP, the DOE includes more stringent requirements such as the following:

1. 10 CFR 835.101(a) requires approval by DOE.

2. 10 CFR 835.101(b) gives authority to DOE to direct or make modifications to a RPP.

3. 10 CFR 835.101(e) requires the RPP to address, but not necessarily be limited to, each requirement in Part 835.

The NRC requires the development, documentation, and implementation of a radiation protection program by its licensees (10 CFR 20.1101(a)), but the regulation does not require the licensee to submit this document for approval by the Agency, and does not give the NRC the direct authority to require modifications of the RPP.

The relevant sections of 10 CFR 20.1101 and 10 CFR 835.101 are provided in Table 15. This table indicates that there are few corresponding sections in the two Parts. There are similarities between 10 CFR 20.1101(a) and 10 CFR 835.101(a) and also between 10 CFR 20.1101(c) and 10 CFR 835.101(g), but, otherwise, the two regulations are quite different. The related recordkeeping requirements of 10 CFR 20.2102 have some similarities to portions of 10 CFR 835.701 and 10 CFR 835.704, as shown in Table 16.

Table 15: Radiation Protection Plan Requirements

Part 20	Part 835
Subpart B - Radiation Protection Programs	**Subpart B - Management and Administrative Requirements**
10 CFR 20.1101 Radiation protection programs	**10 CFR 835.101 Radiation protection programs**
(a) Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part. (See 10 CFR20.2102 for record- keeping requirements relating to these programs.)	(a) A DOE activity shall be conducted in compliance with a documented radiation protection program (RPP) as approved by the DOE.
(b) The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).	*(no corresponding regulation)*
(no corresponding regulation)	(b) The DOE may direct or make modifications to a RPP.
(no corresponding regulation)	(c) The content of each RPP shall be commensurate with the nature of the activities performed and shall include formal plans and measures for applying the as low as reasonably achievable (ALARA) process to occupational exposure.
(no corresponding regulation)	(d) The RPP shall specify the existing and/or anticipated operational tasks that are intended to be within the scope of the RPP. Except as provided in Sec. 835.101(h), any task outside the scope of a RPP shall not be initiated until an update of the RPP is approved by DOE.
(no corresponding regulation)	(e) The content of the RPP shall address, but shall not necessarily be limited to, each requirement in this part.
(no corresponding regulation)	(f) The RPP shall include plans, schedules, and other measures for achieving compliance with regulations of this part. Unless otherwise specified in this part, compliance with amendments to this part shall be achieved no later than 180 days following approval of the revised RPP by DOE. Compliance with the requirements of Sec. 835.402(d) for radiobio- assay program accreditation shall be achieved no later than January 1, 2002.
(c) The licensee shall periodically (at least annually) review the radiation protection program content and implementation.	(g) An update of the RPP shall be submitted to DOE: (1) Whenever a change or an addition to the RPP is made; (2) Prior to the initiation of a task not within the scope of the RPP; or (3) Within 180 days of the effective date of any modifications to this part.
(no corresponding regulation)	(h) Changes, additions, or updates to the RPP may become effective without prior Department approval only if the changes do not decrease the effectiveness of the RPP and the RPP, as changed, continues to meet the requirements of this part. Proposed changes that decrease the effectiveness of the RPP shall not be implemented without submittal to and approval by the Department.
(no corresponding regulation)	(i) An initial RPP or an update shall be considered approved 180 days after its submission unless rejected by DOE at an earlier date.

Table 16: Recordkeeping Requirements

Part 20	Part 835
Subpart L - Records	**Subpart H - Records**
10 CFR 20.2102 Records of Radiation Protection programs	**10 CFR 835.701 General Provisions**
(a) Each licensee shall maintain records of the radiation protection program, including: (1) The provisions of the program; and (2) Audits and other reviews of program content and implementation. (b) The licensee shall retain the records required by paragraph (a)(1) of this section until the Commission terminates each pertinent license requiring the record. The licensee shall retain the records required by paragraph (a)(2) of this section for 3 years after the record is made.	(a) Records shall be maintained to document compliance with this part and with radiation protection programs required by 10 CFR 835.101. (b) Unless otherwise specified in this subpart, records shall be retained until final disposition is authorized by DOE.
	10 CFR 835.704 Administrative records
(no corresponding regulation)	(b) Actions taken to maintain occupational exposures as low as reasonably achievable, including the actions required for this purpose by Sec. 835.101, as well as facility design and control actions required by Secs. 835.1001, 835.1002, and 835.1003, shall be documented.
See 10 CFR 20.2102(a)(2), above.	(c) Records shall be maintained to document the results of internal audits and other reviews of program content and implementation.

As stated previously, the 10 CFR Part 20 (NRC) requirements for an applicant's or licensee's RPP are less prescriptive than the 10 CFR Part 835 (DOE) requirements. If regulatory authority for the TWRS-P Project were to transition from DOE to NRC, in the process of replacing references to DOE and 10 CFR Part 835 with references to NRC and 10 CFR Parts 20 and 70, the contractor may want to use the opportunity to remove requirements that are not part of the NRC regulations. This could be problematical, given that changes that are judged to reduce the effectiveness of the RPP may not be acceptable.

If the NRC were to assume regulatory authority at an early stage, i.e., prior to construction and/or operation, the content and format of the RPP would be reviewed in the context of a Preliminary/Final Safety Analysis Report (or equivalent), in accordance with the NRC's Standard Review Plan (SRP - see Section 3.4.2.6). In any case, the contractor's RPP would be reviewed within the scope of the NRC's inspection program to determine whether it is in compliance with 10 CFR 20.1101a, "Radiation protection programs," which states:

> Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part.

For the TWRS-P facility, in addition to the categories of general employee and radiological worker (defined in Part 835) and member of the public (defined both in Part 835 and in Part 20), a third category known as the co-located worker (CLW), has been defined within the "Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization

Contractors," DOE/RL-96-0006, Revision 1, July 1998. Within this document, the co-located worker is defined in Section 6.0, Glossary, 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.

The co-located worker is referenced in the Introduction to Section IV of the DOE Implementation Guide G-10 CFR 835/B2, "Occupational ALARA Program" (DOE 1994), which states:

> This section gives the basic guidelines for conducting an operational ALARA program. It includes the requirements and guidance for developing, implementing, and documenting the program to reduce doses to workers and "co-located workers" to levels that are ALARA.

Currently no plans exist regarding the change in status, if any, of the CLW category if a transition to NRC regulatory authority were to occur.

3.4.2.4. ORGANIZATION OF THE RPP FOR DESIGN

The first four sections of the RPP are the "Introduction," "RPP Document Organization," "Purpose," and "Applicability." Most of the relevant information in these sections has been summarized in the introduction to this section (3.4.2). The major content of the RPP is contained in RPP Section 5, which addresses "... each functional element of 10 CFR Part 835 applicable during the RPP-WTP design phase" (Section 5.0, Paragraph 4). RPP Section 5.1 covers "Maintenance of the RPP." Sections 5.2, 5.3, and 5.4 discuss Audits, Management of Records, and Training, respectively, which are required to demonstrate compliance with 10 CFR Part 835.

RPP Section 5.5, "BNFL Inc. Application of ALARA to the RPP-WTP Facility Design," is comprised of 14 subsections that are intended to describe how the contractor will apply ALARA to the facility design. Two of the fourteen RPP subsections were deemed inapplicable to the design phase: 5.5.4, "Radiological Performance Goals/Indicators," and 5.5.13, "Radiological Work/Experiment Planning."

RPP Subsections 5.5.1 through 5.5.14 are summarized in Table 17. These are based on the essential elements in Section IV of the DOE Implementation Guide G-10 CFR 835/B2, "Occupational ALARA Program" (DOE 1994). The essential element numbers are shown in **boldface** type below the corresponding subsection numbers.

The first paragraph of RPP Section 5.5 states "the form and content of this section of the RPP is consistent with" the DOE Implementation Guide. The information in this Implementation Guide is similar to that found in NRC Regulatory Guides 8.8 and 8.10. Although a comprehensive intercomparison of regulatory guidance is not the intention of this summary, the following chapter provides an indication of similarities and differences of approach between the NRC and DOE.

The RPP for Design concludes with a Bibliography (Section 6.0) and an Appendix of nearly 40 pages that lists in a tabular format each section of Part 835 and its requirements, accompanied by the BNFL Inc. Plans and Measures for Achieving Compliance. If a particular section of Part 835 is not applicable at the design stage, or if no actual requirement is stated, this is so indicated in the table. Most plans and measures are described by reference to the appropriate part of the text of the RPP.

Table 17: Summary of Application of ALARA to the Facility Design

RPP Section	Section Title	Summary of Section Contents
5.5.	BNFL Inc. Application of ALARA to the Facility Design	
5.5.1. **(1)***	ALARA Policy/ Management Commitment	Policy is to conduct radiological operations in a manner that ensures the health and safety of all employees, subcontractors, and the general public.
5.5.2. **(2)**	Organization and Responsibilities	Project General Manager Project Manager Project Engineering Manager Project Quality Assurance Manager Project Operations Manager (Design Phase) Project Safety and Regulatory Programs Manager Project Safety Committee ALARA Subcommittee (ASC)
5.5.3. **(3)**	Administrative Control Levels	"Administrative Control Levels," are described in DOE Implementation Guide G-10 CFR Part 835/B2, "Occupational ALARA Program" (DOE 1994) in relation to occupational exposure and are, therefore, not applicable to this RPP.
5.5.4. **(4)**	Radiological Performance Goals/ Indicators	This RPP reflects the design phase of the Project only. Radiological Performance Goals/Indicators are not applicable to this RPP.
5.5.5. **(5)**	ALARA Training	Specific technical training (including ALARA training) shall be planned, scheduled, provided, documented, and maintained for personnel in their respective technical disciplines as defined by position descriptions and specific work assignments. A training matrix shall be maintained for defining and tracking training requirements.
5.5.6. **(6)**	Plans and Procedures	Plans and procedures for ALARA shall be commensurate with the activities authorized to be performed under this RPP.
5.5.7. **(7)**	Internal Assessments/ Audits	Perform internal audits of all functional elements of the radiological control program. Such audits shall ensure that each functional element, including ALARA, is audited formally no less frequently than every 3 years.

*Numbers in **boldface** type refer to the corresponding essential elements in DOE IG G-10 CFR Part 835/B2, Occupational ALARA Program

Table 17: Summary of Application of ALARA to the Facility Design (continued)

RPP Section	Section Title	Summary of Section Contents
5.5.8. **(8)**	Optimization Methodology	
5.5.8.1.	Process Description	It is not expected that cost benefit analysis (CBA) will be used as the primary driver in every ALARA decision. CBA is viewed as one of the inputs in an ALARA analysis, and depending on the particulars of the situation (the formality and degree of quantitative analysis should reflect the scale and type of problem under consideration), CBA may not be required in order to arrive at an appropriate decision.
5.5.8.2.	Applied Value of Protection for Optimization	A key input parameter for CBA is the value of the person-rem detriment. BNFL Inc. policy is based on the United Kingdom National Radiation Protection Board (NRPB) guidance, together with an adjustment for United States commercial and DOE practices.
5.5.9. **(9)**	ALARA Design Process	
5.5.9.1.	Overview	The ALARA design process applied in all stages of design. Experience has shown that the greatest potential for significant dose savings at the lowest cost is achieved at the earliest stages of design.
5.5.9.2.	Hierarchy of Protection	ALARA design process uses a hierarchy of controls giving priority to those controls that are most effective.
5.5.10.	ALARA Design Criteria	ALARA design criteria applied throughout the design of the facility: 5.5.10.1. Primary method - physical design features (e.g., confinement, ventilation, remote handling, and shielding). 5.5.10.2. Administrative controls - employed only as supplemental methods to control radiation exposure. 5.5.10.3. Physical design features for specific activities demonstrated to be impractical - use administrative controls. 5.5.10.4. Optimization methods (i.e., cost benefit analyses) - to assure that occupational exposure is maintained ALARA in developing and justifying facility design and physical controls. (Not used, or documented in all ALARA decisions.) 5.5.10.5. Design objective, under normal condition - avoid releases of airborne radioactive materials to the workplace atmosphere. 5.5.10.6. Confinement and ventilation - control the inhalation of airborne radioactive material by workers to levels that are ALARA. 5.5.10.7. Design or modification of a facility and the selection of materials - include features that facilitate operations, maintenance, decontamination, and decommissioning. 5.5.10.8. Combination of physical design features and administrative controls - provide that anticipated occupational dose not exceed limits in 10 CFR 835.202; ALARA process is utilized for personnel exposures to ionizing radiation. 5.5.10.9. Design objective for external sources of radiation in areas of continuous occupancy (2,000 hours per year) - maintain exposure levels below average of 0.5 mrem per hour and as far below this average as is reasonably achievable. 5.5.10.10 Design objectives where occupancy differs from the above (e.g., less than 2,000 hours per year) - ALARA and less than 20 percent of the applicable standards in 10 CFR 835.20.

*Numbers in **boldface** type refer to the corresponding essential elements in DOE IG G-10 CFR Part 835/B2, Occupational ALARA Program

Table 17: Summary of Application of ALARA to the Facility Design (continued)

RPP Section	Section Title	Summary of Section Contents
5.5.11.	ALARA Design Process Components	ALARA design process described in procedures and codes of practice approved by Project management. ALARA design process consists of: completion of a baseline design proposal; Identification, evaluation of alternatives for baseline case doses; ALARA assessments; formal ALARA reviews; final decision process; incorporate changes into design; ALARA documentation.
5.5.11.1.	Baseline Design Proposal	Baseline design proposal is a proposed facility, or portion of the facility, that meets the criteria outlined in Section 5.5.10, ALARA Design Criteria.
5.5.11.2.	Identification of Alternatives	For each radiation exposure scenario evaluated during ALARA process, alternatives are generated for later evaluation - ensures alternatives are considered systematically and consistently. · Substitution or minimization of source terms affecting personnel dose · Increased reliability of processes and equipment · Increasing distance and shielding to the source term · Increasing effectiveness of engineered controls · Decreasing the need for exposure · Decreasing exposure time · Modification of the facility layout or process flow.
5.5.11.3.	ALARA Design Assessments	ALARA design assessments are conducted and documented for each part of the design. The following phases or components of the design should be assessed by the designer, either individually or in combination, with involvement by a radiological engineer, as appropriate: · Process · Operation and maintenance philosophy · Plant layout (to include adequate provisions for access and egress to controlled areas, and adequacy of plant monitoring) · Cell layouts · Source minimization · Contamination control · Individual shield items (e.g., glovebox shielding, shield doors, shield windows) · Bulk shielding (walls, ceilings, and floors) · Construction/installation · Design aspects of operation · Design aspects of decommissioning. ALARA assessment normally conducted during site selection. but site has been pre-selected by DOE. Estimate of the dose resulting from each design alternative and the associated cost are needed for ALARA assessment. More than one alternative applied to an exposure situation may provide equivalent ALARA benefit - consider operational experience of existing plants if reasonable to do so. In the general case, in ALARA assessments consider any design modification where: · Reducing dose might result in an increase of a conventional hazard (e.g., risk of injury from collision with equipment). · Result is greater design, construction, operating, or decommissioning costs. · Difficulties in building, operating, or decommissioning the plant are increased. The creation of an additional hazard does not necessarily eliminate selection of an alternative under consideration.

Table 17: Summary of Application of ALARA to the Facility Design (continued)

RPP Section	Section Title	Summary of Section Contents
5.5.11.4.	ALARA Design Reviews	Formal reviews to look for improvements required to demonstrate ALARA compliance and to record ALARA decisions. ALARA reviews also can be used to record where dose reduction has been achieved by the use of "good engineering practices." Reviews should use appropriate checklists to ensure consistency; shall be conducted by personnel not involved directly in producing the design. The outcome of the reviews will record the key ALARA decisions made in each design stage.
5.5.11.5.	Consensus Approval	More than one alternative may be proposed that achieves ALARA objective. ALARA Subcomittee (ASC) will select the optimum alternative. ASC provides recommendation(s) through Plant Safety Committee to Project Manager for consideration and approval. Contested issues should be clearly identified, characterized, and negotiated between ASC and Project Manager. May require General Manager to make final decision.
5.5.11.6.	Incorporate Changes into Design	Following a decision to incorporate the ALARA changes into the design; the changes will be implemented using authorized design change control procedures.
5.5.12.	ALARA Documentation	All records pertaining to the ALARA design review process including formal ALARA design reviews, cost/benefit reviews, design process audits, and assessments that include ALARA shall be retained in accordance with BNFL Inc. records retention procedures.
5.5.13. **(10)**	Radiological Work/Experiment Planning	Because this RPP submittal is for design work only, no element of radiological work or experiment planning need be addressed at this time.
5.5.14. **(11)**	Records	BNFL Inc. shall generate and retain all records necessary to demonstrate compliance with 10 CFR 835.1001. These records include ALARA training records, formal ALARA design reviews, cost/benefit reviews, design process audits, and assessments that include ALARA.

*Numbers in **boldface** type refer to the corresponding essential elements in DOE IG G-10 CFR 10 CFR 835/B2, Occupational ALARA Program

3.4.2.5. COMPARISON OF NRC AND DOE ALARA GUIDANCE

Section 5.5, Paragraph 1 of the RPP for Design, Rev. 3, states "the form and content of this section of the PP [protection program (i.e., RPP)] is consistent with DOE Implementation Guide (IG) G-10 CFR Part 835/B2...," which was issued in November 1994. It was based on the then-current (1993) version of 10 CFR Part 835. Following the 1998 revisions to 10 CFR Part 835, the IG was updated and renumbered as DOE G 441.1-2, which was released on March 17, 1999. However, the contractor may chose to use the latest revision or to continue using the old version. If the contractor chooses to use the old version, it must ascertain that continued use of that version does not conflict with the current (1998) version of 10 CFR Part 835; in other words, compliance with the current law is required. It is also necessary for the contractor to ensure that use of the older version does not reduce the effectiveness of the RPP. At the time this summary was being prepared, the contractor had not conveyed any decision to DOE regarding the use of the revised IG.[36]

There are many differences between the NRC Regulatory Guides (RG) and the 1994 version of the DOE Implementation Guide (IG). The approaches are similar, as would be expected given the similarities in 10 CFR Parts 20 and 835, and given that the development of the ALARA philosophy, under the auspices of the International Commission on Radiological Protection, has served as a common resource for guidance developed by both the NRC and DOE. The Introduction to NRC RG 8.8 states that "[t]his guide provides information relevant to attaining goals and objectives for planning, designing, constructing, operating, and decommissioning a light-water reactor nuclear power station ..." to achieve ALARA. In contrast, RG 8.10, besides applying to all specific licensees, "...describes an operating philosophy that the NRC staff believes all specific licensees should follow to keep occupational exposures to radiation ...[ALARA]." The DOE IG on ALARA applies to all radiological activities within DOE and its various facilities and also takes into account other hazards (e.g., chemical) that workers may encounter. Because RG 8.10 applies to all specific licensees, it is the most directly comparable to the DOE IG. Some topics not covered specifically in RG 8.10 are presented in RG 8.8; however, some material in the DOE IG is not specifically dealt with in either NRC RG. Although RG 8.8 applies specifically to light-water reactor nuclear power stations, the ALARA principles described in it are not exclusive to that type of facility.

Section IV of Implementation Guide G-10 CFR Part 835/B2 provides "the basic guidelines for conducting an operational ALARA program." It lists 11 "...'essential elements' that shall be incorporated into an acceptable occupational ALARA program." There are parallels to this guidance in NRC RGs 8.8 and 8.10. Table 18 lists these to the extent that comparable concepts can be identified. While a direct comparison of the DOE and NRC guidance documents is somewhat limited by the considerable difference in styles in which they are written, the 11 essential elements are as follows. The information in parentheses indicates the primary source of the requirements: DOE's "Radiological Control Manual" (RCM), with the associated numbers denoting the article numbers, and 10 CFR Part 835.

1. Management Commitment: Establish commitment and participation of all line management and all levels of the work force to the ALARA policy (RCM 121);

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36 Per e-mail from Jeanie Polehn, DOE-RU, 3/16/00.

2. Assignment of Responsibilities: Assign specific responsibilities to line management and workers involved in implementing the ALARA program;

3. Administrative Control Levels (RCM 211.2 and 211.4): Establish a challenging level of administrative control that is more restrictive than the DOE RCM Administrative Control Level of 2,000 mrem per year;

4. Radiological Performance Goals (RCM 121.9, 132.1 and 133): Establish, approve, and review quarterly a program of radiological performance goals;

5. ALARA Training: Require the attendance of managers and workers involved with any aspect of radiological operations (RCM 651-654);

6. Plans and Procedures: Set up formal plans and procedures to attain and maintain occupational exposures ALARA (10 CFR 835.101(c));

7. Internal Audits/Assessments: Conduct comprehensive internal reviews, audits, and evaluations periodically and report the results to the highest levels of site management (10 CFR 835.102);

8. Optimization Methodology: Use methods of optimization to assure that occupational exposure is maintained ALARA when developing and justifying the facility design and physical controls during the design of new facilities or major modification of old facilities (10 CFR 835.1002(a));

9. Radiological Design Review: Ensure the integration of appropriate methods and considerations during the design phase to maintain occupational exposures ALARA during subsequent construction, modification, and operation of the equipment or facility (10 CFR 835.1001(a));

10. Radiological Work/Experiment Planning: Integrate measures and controls to maintain occupational exposures ALARA for specific operations and experiments; and

11. Records: Maintain documents that demonstrate compliance and that the program is adequately carried out (10 CFR 835.704(b) and RCM 711-713).

3.4.2.6. RADIATION PROTECTION ASPECTS FROM THE NRC STANDARD REVIEW PLAN

NUREG-1702, "Standard Review Plan for the Review of a License Application for the Tank Waste Remediation System-Privatization (TWRS-P) Project," (Main Reference 15) provides guidance for the review and evaluation of health, safety, and environmental protection in applications for licenses to possess and use special nuclear material (SNM) during the remediation of radioactive tank waste at Hanford. The guidance is also applicable to the review and evaluation of proposed amendments and license renewal applications. Specific filing requirements for license applications and for issuance of such licenses are in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

Table 18: Comparison of DOE Implementation Guide G-10 CFR Part 835/B-2 and NRC Regulatory Guide 8.10

DOE IG G-10 CFR Part 835/B-2 Essential Element **(Number) Title / Narrative Description**	**NRC RG 8.10 or 8.8 Regulatory Position**[37]	**NRC Regulatory Position Description** (where available) Note: most are paraphrased
(1) Management Commitment Establish commitment and participation of all line management and all levels of the work force to the ALARA policy (RCM 121)	8.10 C.1.a.	Plant personnel should be made aware of management's commitment to keep occupational exposures ALARA.
	8.8 C.1.a.	Program for Maintaining Station Personnel Radiation Doses ALARA.
(2) Assignment of Responsibilities Assign specific responsibilities to line management and workers involved in implementing the ALARA program	8.10 C.1.c.	Ensure well-supervised radiation protection capability with well-defined responsibilities.
	8.8 C.1.b.	Organization, Personnel, and Responsibilities.
(3) Administrative Control Levels Establish a challenging level of administrative control that is more restrictive than the DOE RCM Administrative Control Level of 2,000 mrem per year	*No direct comparison*	A similar goal is sought under 8.10.C.1.f. (Modifications to procedures/equipment/facilities should be made where they will substantially reduce exposures at a reasonable cost) and under 8.10.C.2.b. (Periodically review procedures to identify how exposures can be reduced).
(4) Radiological Performance Goals Establish, approve, and review quarterly a program of radiological performance goals;	*No direct comparison*	The DOE IG requires a radiological performance goal program, to be reviewed quarterly. RG 8.8 C.1.b(1)(a)-(e) discusses a corporate program with specific objectives and periodic review.
(5) ALARA Training Require the attendance of managers and workers involved with any aspect of radiological operations	8.10 C.1.d.	Plant workers receive sufficient training.
	8.8 C.1.c.	Discusses who should receive training, and to what extent; also retraining; minimum content.
(6) Plans and Procedures Set up formal plans and procedures to attain and maintain occupational exposures ALARA	8.10 C.1.e., C.1.f., C.2.b.	RSO should be given sufficient authority; Modifications to reduce exposures should be made; RSO & staff look for ways to reduce exposure.
(7) Internal Audits/Assessments Conduct comprehensive internal reviews, audits, and evaluations periodically and report the results to the highest levels of site management	8.10 C.1.b.	Periodically perform a formal audit to determine how exposure might be lowered.

[37] R.G. 8.10 is listed first because it is the more general and, therefore, more directly comparable to the DOE IG.

Table 18: Comparison of DOE Implementation Guide G-10 CFR Part 835/B-2 and NRC Regulatory Guide 8.10 (continued)

DOE IG G-10 CFR 835/B-2 Essential Element **(Number) Title / Narrative Description**	**NRC RG 8.10 or 8.8 Regulatory Position**[37]	**NRC Regulatory Position Description** (where available) Note: most are paraphrased
(8) Optimization Methodology Use methods of optimization to assure that occupational exposure is maintained ALARA when developing and justifying the facility design and physical controls during the design of new facilities or major modification of old facilities	*No direct comparison*	The DOE IG indicates optimization methodology shall be used per 10 CFR 835.1002(a) and refers to a formally documented optimization methodology for certain sites. No identical NRC requirement.
(9) Radiological Design Review Ensure the integration of appropriate methods and considerations during the design phase to maintain occupational exposures ALARA during subsequent construction, modification, and operation of the equipment or facility	8.10 C.1.f.	Modifications to reduce exposures should be made.
	8.8 C.1.d	Review of New or Modified Designs and Equipment Selection.
(10) Radiological Work/Experiment Planning Integrate measures and controls to maintain occupational exposures ALARA for specific operations and experiments	8.8.C.3.a. & b.	The DOE IG specifies a formal ALARA review should be carried out, while the NRC RG 8.8. C.3.a. & b. provide similar guidance, though less prescriptive.
(11) Records Maintain documents that demonstrate compliance and that the program is adequately carried out	*No direct comparison*	RG 8.8 C.3 discusses recordkeeping within a radiation protection program, but is less prescriptive than the DOE IG.

The purposes of the SRP are to:

1. Ensure the quality and uniformity of staff reviews.
2. Present a well-defined base from which to evaluate proposed changes in the scope, level of detail, and acceptance criteria of reviews.
3. Serve as the basis for the review of requests by licensees for changes in their licenses.

Thus, the SRP, at any point in time, can provide a basis for the review of proposed new or renewal applications and amendments to existing licenses, as well as modifications to the SRP resulting from new NRC requirements and licensee initiatives.

[37] R.G. 8.10 is listed first because it is the more general and, therefore, more directly comparable to the DOE IG.

The responsibility of the staff in review of license application, renewal application, or license amendment for a TWRS-P facility is to determine that:

1. There is reasonable assurance that the facility can and will be operated in a manner that will not be inimical to the common defense and security.

2. The facility will provide reasonable protection of the health and safety of workers, the public, and the environment.

To carry out this responsibility, the staff:

1. Evaluates information provided by an applicant.

2. Determines, through independent assessments, that the applicant has demonstrated a reasonable safety program that is in accordance with regulatory requirements.

To facilitate carrying out this responsibility, the SRP clearly states and identifies those standards, criteria, and bases that the staff should use in reaching licensing decisions.

Although 10 CFR Part 70, as revised[38], does not specifically include a TWRS-P facility in its list of activities requiring the inclusion of requirements found in Subpart H of 10 CFR Part 70, the staff believes that a TWRS-P facility is an activity that could significantly affect public health and safety and, therefore, plans to invoke the requirements found in Subpart H for this type of facility. As such, NRC requirements in 10 CFR Part 70, as revised, require that an applicant submit a complete description of the safety program for the possession and use of SNM to show how compliance with the applicable requirements will be accomplished.

The requirements in 10 CFR Part 70 specify, in general terms, the information to be supplied in a Safety Program Description. The specific information to be submitted by an applicant and evaluated by staff is identified in the SRP. A license application should contain a Safety Program Description that addresses all the topics in the "Table of Contents" of the SRP.

The applicant's, or licensee's, Radiation Safety Program and Radiation Safety Design Features are covered in Chapters 4.1 and 4.2 of NUREG-1702, respectively. These chapters contain specific information concerning what constitutes an acceptable Radiation Protection Program, from the perspective of the NRC. The 11 essential elements of the DOE G-10 CFR Part 835/B2, shown in Table 18, are not expressly cited in the SRP; however, the guidance contains provisions for the review of ALARA considerations (at the design stage, during operations, and for modifications), organizational relationships and personnel qualifications, and training. These are also covered in both the DOE IG and the NRC RGs. The remaining items in Table 3.4.2.4 would all be considered parts of a good radiation protection program, as evidenced by the similarities to guidance found in the NRC RGs 8.8 and 8.10. These items would not be explicitly covered by the SRP-mandated review, but this should not be considered as a justification for eliminating any existing elements of the RPP and, therefore, possibly reducing its effectiveness.

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[38] This reference is to the draft revision to 10 CFR Part 70, subject to on-going dialogue. The SRP uses sidebars to indicate additional references to the draft version of 10 CFR Part 70.

3.4.2.7. SUMMARY

This summary has provided a discussion of the RPP for the Tank Waste Remediation System-Privatization Project, concentrating on differences in regulatory requirements and guidance of DOE and the NRC. At the time this summary was being prepared, the contractor had not yet produced an acceptable version of the RPP for Design and Limited Construction; therefore, this discussion has focused on Rev. 3, the RPP for Design.

It was noted that the NRC requires the development, documentation, and implementation of an RPP by its licensees (10 CFR 20.1101(a)), but it does not require the licensee to submit this document for approval by the Agency and does not give it the direct authority to require modifications of the RPP. By way of contrast, the regulations in 10 CFR 835.101 are very prescriptive with respect to the requirement for an RPP. While both agencies require the implementation of an RPP, the DOE includes more stringent requirements such as approval by DOE, authority of DOE to direct or make modifications to an RPP, and the requirement that the RPP must address each requirement in 10 CFR Part 835.

This summary also included comparisons of the guidance documents issued by the two agencies that are intended to assist the regulated entities in understanding and complying with its regulations (10 CFR Parts 20 and 835). As noted in Section 3.4.2.5, direct comparison of the DOE and NRC guidance documents is somewhat limited by the difference in styles in which they are written, as well as the difference in orientation (e.g., types of facilities, regulatory basis) and emphasis (as shown in Table 18). The DOE IG G-10 CFR Part 835/B2 lists 11 "essential elements" that are required for an acceptable occupational ALARA program. The NRC RGs are not so specific.

The TWRS-P facility would be subject to review in accordance with the NRC's SRP, NUREG-1702, if there were a transition of regulatory authority from DOE to the NRC. Section 3.4.2.6 indicates how the SRP would relate to the contractor's RPP. It was stated there that the SRP, at any point in time, can provide a basis for the review of proposed new or renewal applications, and amendments to existing licenses, as well as modifications to the SRP resulting from new NRC requirements and licensee initiatives. This does not necessarily imply that an existing facility would be required to make modifications as transition to NRC regulatory authority takes place. However, if regulated by the NRC, future modifications and license renewal applications would likely be reviewed using the SRP.

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3.0 AREAS OF REVIEW
3.5 CRITICALITY SAFETY

3.5.1 CRITICALITY SAFETY PROGRAM DOCUMENT AND REQUIREMENTS

The umbrella Nuclear Criticality Safety program document, "Criticality Safety Program for the River Protection Project–Waste Treatment Plant, PL-W375-NS00001, Revision 1" provides an overall description of the Nuclear Criticality Safety program. The intent of this program document is to implement the U.S. Department of Energy (DOE) Top-Level requirement in DOE/RL-96-0006: "The facility shall be designed and operated in a manner that prevents nuclear criticality." While the document adheres to good practices generally accepted in the criticality safety community as well as the guidance provided by American National Standard Institute/American Nuclear Society (ANSI/ANS) 8.1," Nuclear Criticality Safety in Operations with Fissionable Materials outside Reactors," and ANSI/ANS-8.19, "Administrative Practices for Nuclear Criticality Safety," it lacks sufficient detail with respect to implementation of the Nuclear Criticality Safety Program and the controls to be used at the proposed new waste treatment facility.

There is a commitment to adhere to the Double Contingency Principle. Section 5.1 of PL-W375-NS00001states, "Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." The document goes on to say that Double Contingency protection can be provided by either the control of two independent process parameters (which is the preferred approach, when practical, to prevent common-mode failure) OR by a system of multiple controls on a single parameter. Depending on the specifics of the situation, multiple controls on a single parameter may not be sufficient to demonstrate double contingency to the U.S. Nuclear Regulatory Commission (NRC).

Appendix 1 identifies work which will need to be completed to resolve open technical issues. These issues include parametric studies of k_{eff} for worst case waste mixtures as opposed to the Pu-water mixture discussed in the Initial Safety Analysis Report, study fissile material over-concentration accidents in the melter, study criticality parameters for the low activity waste process train, study solids settling in process vessels, and study boiling and drying-out of waste due to excess temperature. The River Protection Project-Waste Treatment Plant (RPP-WTP) has committed to initiate these activities prior to the submittal of the Preliminary Safety Analysis Report and complete these activities prior to submission of the Final Safety Analysis Report. However, some of these technical issues may be difficult to resolve. It is essential to identify any technical issues that could potentially delay start-up of the facility as early in the process as possible to allow for adequate time to study and resolve such issues.

3.5.2 INTERIM CRITICALITY SAFETY EVALUATIONS

Based on the interim criticality evaluations submitted, BNFL Inc. developed its nuclear criticality safety program by bounding k_{eff} for nearly dry waste solids with a maximum plutonium content. This approach relies upon providing controls on the feed and analyzing the process to ensure that no normal operations or process upsets could result in formation of a critical mass. As in the Criticality Safety Program document, double contingency is required but is permitted by either the control of two independent process parameters (which is the preferred approach, when practical, to prevent common-mode failure) OR by a system of multiple controls on a single parameter. As indicated above, multiple controls on a single parameter may not be sufficient to demonstrate double contingency to the NRC. Additionally, there is not sufficient detail to determine how double contingency will be applied to the processes nor is it clear that all potential concentration mechanisms (e.g. preferential settling of specific isotopes) have been evaluated.

In the evaluations reviewed, BNFL Inc. has relied upon concentration limits for special nuclear material as provided for in their contract with DOE as an upper boundary for criticality analyses, despite the fact that there are waste forms within the Hanford tank waste that may be above those limits. The criticality values are from the Carter Handbook that is widely used in the DOE complex. However, these have not undergone rigorous validation. The NRC uses ANSI/ANS 8.1 and other ANS standards that have undergone rigorous validation as detailed in Regulatory Guide 3.71. Also, BNFL Inc. has committed to only one criticality safety alarm in each area requiring alarms which is inconsistent with 10 CFR Part 70 requirements which requires that "coverage of all areas shall be provided by two detectors." This is one situation where the NRC guidance is more restrictive than the ANSI standard which allows coverage by one detector.

There are processes or potential processes and operations that could occur at the facility that have not been discussed in any criticality safety evaluations such as solvent extraction and precipitation of solids which could have significant safety concerns, especially with the uncertainties in the waste stream.

3.5.3 UNCERTAINTIES AND UNRESOLVED NUCLEAR CRITICALITY SAFETY ISSUES

In addition to the unresolved technical issues that are identified in the nuclear criticality safety (NCS) program document, there are additional technical items identified in the limited number of criticality safety evaluations submitted to the NRC, such as studying the possibility of solids precipitation from the mixing of fissile streams, verifying the maximum density of solids and verifying the maximum fissile concentration in the glass product. A major point of concern is the large amount of uncertainty in the waste tank feed material. It seems extremely difficult to determine the "worst-case " conditions for analysis when there is so much question as to the make-up of the input waste stream. Especially when considering the possibility of build-up on resin (such as ion exchange) filters and processes like solvent extraction and precipitation where fissile material could reach potentially high concentrations, great variability in the input waste stream could lead to large differences in process conditions.

3.5.4 CONFORMANCE WITH NRC REGULATIONS

Based on the criticality safety program document and criticality safety analyses provided, more information would be required to determine the overall conformance with NRC regulations. With the exception of the issues addressed above involving meeting the double contingency principle and the issue of dual coverage with respect to criticality alarms (this would require either an exemption or facility changes should the NRC ever assume regulatory authority), it is not possible to determine the state of conformance with NRC regulations for the proposed new vitrification facility. Much more detail about the design of the facility and processes would be required by the NRC to make this assessment. Use of the ANSI/ANS 8.1 subcritical limits is endorsed by the NRC as stated in Regulatory Guide 3.71 and is preferable the approach chosen by BNFL Inc. to use safe concentration limits as established by the Carter Handbook. Additionally, good practices like using engineered controls like safe geometry process vessels and defense-in-depth should be practiced. Based on the information provided, conformance with NRC regulations would certainly be possible, especially if NRC regulations were considered in the early stages of design of the facility and processes. More detail about implementation of the criticality program and specifics with regard to items that would be relied upon for criticality safety and risk estimates would be required to fully assess the state of safety and compliance with respect to applicable NRC criticality safety regulations.

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3.0 AREAS OF REVIEW
3.6 PROCESS AND CHEMICAL SAFETY

3.6.1 STANDARDS APPROVAL PACKAGE (SAFETY REQUIREMENTS DOCUMENT AND HAZARDS ANALYSIS REPORT) AND INITIAL SAFETY ANALYSIS REPORT

The BNFL Inc. Safety Requirements Document (SRD) and Hazards Analysis Report (HAR) arrived in September 1997 and identified numerous process and chemical safety scenarios (many with both radiological and chemical consequences) using a preliminary hazard and operability (HAZOP) method. The Contractor used reviewers with expertise in similar types of plants and processes to bridge areas of incomplete information because the preliminary design was still undergoing development and, as such, many open areas existed. The HAZOP used an American Institute of Chemical Engineers (AIChE) risk matrix (a binning approach) to assist with identifying those potential scenarios that posed a sufficient risk (i.e., consideration of both frequency and consequence) to warrant quantitative risk analysis and the consideration of safety controls. In general, the list included many entries of a benign nature that obscured some of the more hazardous conditions at the proposed facility. In addition, the expert review panel inadvertently included the effects of mitigation in some of their assessments, and, consequently, the frequency and consequence bins for some events were underestimated. Thus, potential items relied on for safety (IROFS) were overlooked. From these reviews, the following issues were identified in the process and chemical safety area:

1. Incomplete process and hazard descriptions.

2. Inconsistency of unmitigated and mitigated analyses in the HAR and fault schedules.

3. Emphasis on active instead of passive mitigation and control.

4. Little information on the chemical storage areas.

5. No consideration of chemical effects due to ammonia, several resin reactions, and nitrogen oxides.

6. Inconsistencies in hazards identification, structures, systems, and components (SSC) categorization, and design classes.

7. Relative lack of quantification.

8. Presence of large tanks and inventories.

9. Limited consideration of offnormal and unanticipated events, and interactions.

The actual comments may be found in the SRD/HAR comment transmittal letter (see Footnote 18 on page 39).

The Initial Safety Analysis Report (ISAR) was dated January 12, 1998. The ISAR represented a more detailed design as compared to the SRD/HAR package. The ISAR advanced the design and presented quantitative analyses on several process and chemical safety scenarios, including tank failures, crystalline silicotitanate powder dispersion, ion exchange resin fires, glass spill, anhydrous ammonia leaks, and a nitric acid spill. However, the level of the design corresponded to a preliminary conceptual to adequately address many of the concerns and issues. Furthermore, the U.S. Nuclear Regulatory Commission (NRC) staff's analyses indicated the input values and methods used appeared to be nonrepresentative and could underestimate the potential consequences. For example, the temperature for the vapor calculations and the chemical inventories (particularly nitric acid) at risk appeared to be low. The following comments and issues were raised in the process and chemical safety arena as a result of the ISAR review:

1. Lack of detailed process information and chemical interaction data.
2. Lack of conservatism.
3. Inconsistency of unmitigated and mitigated analyses.
4. Emphasis on active instead of passive mitigation and control.
5. Little information on chemical storage areas.
6. No IROFS's identified for chemical concerns.
7. Inconsistencies in hazards identification, SSC categorization, and design classes.
8. Relative lack of quantification.
9. Presence of large tanks and inventories.
10. Limited consideration of offnormal and unanticipated events, and interactions.

The actual comments may be found in the ISAR comment transmittal letter (see Footnote 19 on page 40).

3.6.2 DESIGN SAFETY FEATURES SUBMITTAL

The BNFL Inc. Design Safety Features (DSF) submittal consisted of two parts. Part 1 discussed general features used to address general hazards and concerns. In essence, this part presented the practical interpretation of the codes and standards from the SRD. "Standard" chemical process industry approaches would be used unless radiological limits were shown to be exceeded. Part 2 discussed and analyzed ten examples in more detail, including loss of confinement events with both radiological and chemical consequences. The following comments and questions were raised in the process and chemical safety arena from the DSF review:

1. Lack of detailed process information and chemical interaction data.
2. Lack of conservatism and consideration of uncertainties.
3. Consideration of corrosion and materials.
4. Juxtaposition of mitigated and unmitigated analyses.
5. Inconsistencies in the analyses.

The actual comments may be found in the DSF comment transmittal letter (see Footnote 34 on page 64). The DSF constituted an example of approaches for safety analyses and did not represent an actual design or safety document.

3.6.3 RELATED ACTIVITIES AND TOPICAL MEETINGS

The DSF submittal and related subjects were discussed at the Topical Meeting in Richland on March 27, 1999. A follow-up item from the meeting was the U.S. Department of Energy/Regulatory Unit (DOE/RU) recommendation that the Contractor develop an acceptable approach to the treatment of chemical hazards that is consistent with the treatment of radiological hazards, as required by the contract. During a teleconference call, the Contractor stated its intent to use standard commercial practice and process industry standards for chemical safety. It was also indicated that a subcontractor would completely construct the cold chemical storage area(s). The Contractor subsequently submitted a position paper to the RU proposing to relax some of the SRD criteria that would have required the quantitative evaluation of essentially all chemicals at the proposed facility in order to achieve cost savings. The RU did not accept that position[39]. The Contractor maintained that there were essentially no IROFSs related to chemical safety.

The April 2000 Topical Meeting also discussed chemical safety. This meeting generated many issues which are discussed in the next section. Summarizing, the Contractor stated they were following the standard, chemical process industry approach to design and safety, and from their analyses for TWRS-P, this could correspond to worker risk levels of circa 1E-3/yr. It was noted by NRC staff present at the meeting that this could be two orders of magnitude greater than the estimated worker radiological risk at the proposed facility. DOE appeared to be cognizant of the disparity. However, because the DOE/RU regulates by the contract using the approved SRD and the SRD incorporates chemical process industry standards, the RU appeared to be willing to acquiesce to a much higher level of risk from chemical events than from radioactive events.

3.6.4 NRC ASSESSMENT OF THE PRESENT STATUS

The NRC staff developed a point paper on the process safety issues associated with the proposed facility (see Section 3.3 of Chapter 3.0 and Chapter 7.0, Main References 18 and 19). This paper indicated IROFS would likely be needed to reduce the chemical risks to levels commensurate with NRC regulations in 10 CFR Part 70 and to meet the SRP guidance. For example, controls would be needed to maintain plant habitability during potential releases of volatile species (e.g., NOx and ammonia). Other issues from the point paper, the Topical Meeting, and the Topical Meeting documentation include the following general comments and observations:

1. <u>Level of Detail in the Analysis:</u> Several times in the analyses and at the April 2000 Topical Meeting, it was stated that a robust design consistent with chemical industry standards would be used. However, few specific details were presented to substantiate this approach. Limited information was presented on the pneumatic transport system (for glass formers) and its potential hazards. Furthermore, it was mentioned at the Topical Meeting that piping and instrumentation drawings (P&IDs) were not available for the chemical handling area (note, however, that a limited number of process flow diagrams were included in the FFP [Firm Fixed Price] submittal of 4/24/00). It would be

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[39] Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc., December 7, 1999.
Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc., December 22, 1999.

anticipated that the Contractor would provide and use more specific information based upon actual designs and diagrams for the safety analyses.

2. <u>Incomplete and Uneven Application of HAZOP Process:</u> The Contractor does not appear to have conducted a complete HAZOP study of the chemical safety area. In addition, the presence of working components and mitigating controls appears to have been assumed (e.g., the static mixers and different sized connectors) for operational control purposes, which has lead to a lack of identification of potential controls for safety purposes. For example, successful operation of the inline mixers depends upon the pumping mechanisms, flowmeters, other sensors, motor-operated valve/air valves, and control systems.

3. <u>Inappropriate Focus on Routine Operations:</u> The majority of the conditions focus on routine or near routine conditions - sometimes there are areas where SSCs are assumed to work but are omitted from further analysis(e.g., see the previous comment on the inline mixers). Offnormal and accident conditions are not well covered; for example, the plugging of pipes and misfeeds do not appear to have been given adequate consideration. Storage facilities such as these involve considerable operator involvement and activities, with frequent deliveries, and potential misfeeds would likely be anticipated events (e.g., potentially annually). The Contractor's proposed approach of reliance upon different types of hose connectors is likely to become a manual, administrative control because most supply trucks carry multiple connectors of different types. The relative lack of sampling and instrumentation on the chemical compositions allows these misfeeds to propagate through the facility. At least three possible misfeed combinations stand out as having potentially significant hazardous outcomes, and it would be expected that these would be analyzed and discussed in more detail in the report:

   a. Sugar (a white powder) is mistakenly added to the silo holding silica (also a white powder). This misfeed could double or triple the amount of sugar added to the melter, resulting in a loss of redox control and large discharges of NOx and hydrocarbons to the offgas systems.

   b. The full strength nitric acid is sent into the plant and contacts the ion exchange resins, resulting in a resin fire and release.

   c. The concentrated nitric acid is added to the concentrated caustic tank - or vice versa. This would likely overheat the tank contents from the vigorous acid/base neutralization reactions, leading to fuming (NOx) and possibly rupture of the tank (i.e., more NOx if the nitric acid tank is involved).

4. <u>Lack of Consideration of Improved Safety Margins:</u> The Contractor does not appear to have considered simple controls that have the potential to cost effectively address and reduce the risk of some chemical concerns and greatly improve safety margins. For example, inline pH probes would provide immediate feedback of the presence or absence of acids and bases and could be used - in conjunction with density or conductivity measurements - to prevent the unplanned mixing of reactive chemicals.

5. Approach of Chemical Industry Standards: The Contractor indicates the design approach is based upon chemical industry standards. However, as presented, the Contractor's approach does not appear to be consistent with AIChE guidelines for chemical process safety, which generally follow an as low as reasonably achievable (ALARA) like approach for chemicals. For example, the AIChE/Center for Chemical Process Safety (CCPS) Guidelines for Engineering Design for Process Safety (New York, 1993) discusses in detail inherently safer plants for continuing the improvement of the good safety record of the chemical and petrochemical industries (see Chapter 2, for example). These same guidelines emphasize multiple safety layers (redundancy). Neither "inherently safer" nor "multiple safety layers" are evident in the proposed design. Often, the process industry uses a risk matrix that identifies (either qualitatively or quantitatively) the consequences and frequencies of potential events, with the identification of controls to reduce risk as appropriate. This was actually initiated by the Contractor in the original HAR (in 1997), but does not appear to have been pursued further. In addition, best handling practices from the chemical industry, which generally emphasize precautions and redundancy (e.g., for nitric acid), do not appear to have been consulted.

6. Potential High Level of Risk: As presented, the proposed approach has a potential, chemical/process risk level two or more orders of magnitude greater than the radiological risk and one order of magnitude greater than standard worker risk. This appears excessively high. For example, there exists a potential for a multiple fatality accident with a frequency of circa 1E-3/yr from the failure of the LAW off-gas line during a seismic event. This is a potentially significant change from the HAR which dismissed this type of accident as "incredible" (i.e., implied to be less than 1E-6/yr).

7. Potential Inconsistency with NRC Draft Regulations and Guidance: The NRC has issued a standard review plan (SRP) for tank waste remediation system (TWRS) facilities as NUREG-1702, in final form. This identifies a target frequency of 1E-5/yr for accidents of high consequences, and a target frequency range of 1E-2/yr to 1E-5/yr for accidents with intermediate consequences. The application of ALARA (for chemical in addition to radiological events) is implied. Draft NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," and draft NUREG-1701, "Standard Review Plan for the Review of a License Application for the Atomic Vapor Laser Isotope (AVLIS) Facility," have similar guidance. The revision of 10 CFR Part 70 states that accidents with potential high consequences should be rendered highly unlikely by the application of controls (i.e., IROFS), while accidents with potential intermediate consequences should be reduced to unlikely frequencies by the application of controls. Again, the application of ALARA is implied for both radiological and chemical scenarios. High consequences and intermediate consequences refer to potential accidents involving radioactive materials, chemicals evolved by radioactive materials, and chemicals that impact the safe handling of radioactive materials. It would appear that the Contractors handling of some of these potential accidents at a TWRS-Privatization (TWRS-P) facility may not be consistent with the NRC approach. As an example, the release mentioned in Comment #6 would be categorized as a high consequence event and controls would be required to reduce its likelihood if the facility were licensed by NRC.

8. Specific Chemical Characteristics: Specific characteristics of certain chemicals and their associated potential hazards do not appear to have been considered. Chemical compatibility, commingling, storage, handling and other potentially hazardous issues need to be addressed more completely. As cited below, appropriately conservative scenarios and their preventative measures need further assessment:

a. Interactions between concentrated nitric acid and concentrated sodium hydroxide.

b. Process safety and control - example in report of inadvertent transfer of concentrated nitric acid into cesium ion exchange columns.

c. Sucrose-related dust explosions.

d. Ammonia release.

e. NOx emissions from the LAW melter offgas treatment system.

f. Seismic analysis results.

g. Design basis event definition.

h. Construction, laboratory, and maintenance chemicals.

i. Start-up simulants.

In addition, industry has codes and practices for handling many of these chemicals which would be expected to influence the design of chemical handling areas; for example:

Piping - American Society of Mechanical Engineers (ASME) B31.3 Category M Fluid Service.

Acid Storage - ASME Boiler and Pressure Vessel Code (B&PVC) Section VIII.

Material - "Materials of Construction for Nitric Acid" ed. by Monitz and Pollock (National Association of Corrosion Engineers, 1986) is a good reference.

Safety Instrumentation Systems (SIS) - American National Standards Institute/Instrument Society of American (ANSI/ISA) S84.01-1996 and Institute of Electrical and Electronics Engineers-603.

US NRC, "Chemical Process Safety at Fuel Cycle Facilities," NUREG-1601.

US NRC, "Guidance on Management Controls/Quality Assurance Requirements for Operations, Chemical Safety, and Fire Protection for Fuel Cycle Facilities," *Federal Register*, Vol. 54, No. 53, March 21, 1989, pages 11590-11598.

9. Management Measures and Corrosion: It is not clear if the Contractor has adequately considered management measures (including surveillance and maintenance), particularly for chronic effects such as corrosion/erosion. The Contractor stated in the April 2000

Topical Meeting documentation that standard process industry approaches will be followed. However, the AIChE "Guidelines" (referenced previously; page 162 et seq) note that corrosion is a major problem in the process industries, with 31 percent of failures due to general corrosion, 24 percent due to stress corrosion cracking, and 10 percent due to pitting. Thus, the stated, process industry-like approach is unlikely to avoid corrosion failures, and it would seem that the Mechanical Integrity (MI) (ref: 2.3.2.8, page 6) is a little bit sketchy. A complete MI covers more than just procedures, maintenance, quality assurance and training. The contractor might want to review the 29 CFR 1910.119 (j) to see what other elements need to be included in the MI program.

10. Environmental Conditions: The analyses for the Topical Meeting used various environmental conditions in some of the considerations. It is not clear if these are adequately representative of the Hanford site. For example, the nitric acid dispersion calculations use a temperature of 95°F, while the building designs in the related 20-month FFP submittal use 113°F (and a -23°F for the minimum) for equipment exposed to the weather. It is not even clear if the 113°F adequately represents the solar effect at the Hanford site. However, the use of the lower temperature (95°F) is likely to underestimate chemical evaporation rates and effects.

11. Inline Mixers: It is unusual to use inline mixing for dilution and potential mixing reactions with significant enthalpies and in a batch mode (e.g., acid-base interactions). Also, inline mixers place considerable emphasis upon the successful operation of the pumping mechanisms, flowmeters, other sensors, MOV/air valves, and control systems for adequate metering and mixing. These other systems do not appear to have been adequately identified, considered, and analyzed by the Contractor.

12. Reliance upon Administrative Controls: The design approach appears to rely extensively on operator actions to prevent or mitigate the effects of chemical hazards. Examples include the sequential dilutions and the evacuation procedures. The normally accepted practice and NRC regulatory emphasis are minimization of the reliance upon administrative controls.

13. Distributed Control System (DCS): The DCS is used to control the filling of numerous tanks and other important functions such as mixing. Conceivably, the DCS can be considered as part of a safety system (safety instrumented system - SIS). It would seem that ANSI/ISA S84.01-1996 "Application of Safety Instrumented Systems for the Process Industries," IEEE-603, or another standard for safety system controls should be considered as applicable to the design of the chemical storage areas.

14. Delivery Schedule and Traffic: It was mentioned during the Topical Meeting that deliveries would be almost continuous during the daylight shift (almost a tank truck every hour). From the presentation, offloading would occur at least 75 percent of the dayshift time for the chemical storage area near the Pretreatment Building and at least 90 percent of the time for the glass formers storage facility. It is not clear if the safety analyses and control strategies have adequately evaluated the potential impact from these frequent deliveries, such as the increased frequencies and inventories for spills and major accidents, and the necessary design and management measures. For example, a tank truck might contain 5,000 gallons of nitric acid in addition to the plant's inventory, and part of the delivery time would be outside the diked areas. This is not currently considered. A truck accident does not appear to have been analyzed. Standard design features for offloading reactive chemicals such as

diked/capture basins and an enclosure with water sprays do not appear to be included in the design. It would be anticipated that a future revision will address these oversights.

The staff has also developed specific comments on the chemical safety approach at the proposed facility based upon the April 2000 Topical meeting and its supporting documentation. Document citations refer to:

Department of Energy (U.S.) (DOE). RTP-W375-SA-00010, Rev. A, "Waste Treatment Plant Chemical Hazards," Draft Report Review Comments by Janet Ledbetter Ferrill and Jim Scott/Div. 01. DOE: Richland, Washington. April 19, 2000.

The specific comments and observations are as follows:

15. Adequacy of Chemical-Only Controls in a Radiochemical Facility: According to Section 2. on "Chemical Hazard Methodology," the "chemical-only " control strategies and standards are based on best industry practice and not on calculated risk. It is not clear if any particular industry or other vitrification facility has been used as precedence for determining these practices. As noted in the text, DOE Regulatory Guide 1.78 for control room habitability during a hazardous chemical release was published 26 years ago (June 1974) and the NUREGs are nuclear based. Other current practice is to prepare Risk Management Program (RMP) per U.S. Environmental Protection Agency (EPA) guidance. It would be anticipated that control strategies would have to address some radiological concerns at the plant caused by chemicals such as plumes/habitability, fires, and misfeeds based upon some form of integrated safety analysis and perception issues. A high-profile radiochemical facility like TWRS-P would be unlikely to accept the accident risk of the process industries even in its chemical handling areas.

16. Properties and Concerns of Ammonia: Ammonia is the number one chemical involved in accidents reported under the EPA's risk management program for high risk industries. The National Institute for Occupational Safety and Health (NIOSH) immediate danger to life and health (IDLH) and Emergency Response Planning Guideline-3 (ERPG) values for ammonia are 300 ppm and 1000 ppm, respectively. In Section 3.3 of the report (page 23) it is stated that 43 lb/hr of ammonia will be going to the selective catalytic reduction unit for NO_x abatement. A break in the ammonia supply line to NO_x abatement system could, therefore, result in a release of around 16.5 cfm of ammonia gas (at room temperature), and would require about 55,000 cfm of air to adequately dilute the ammonia to below the IDLH value of 300 ppm. This accident scenario, which could constitute a significant toxicological hazard to the facility worker, has not been considered in the Waste Treatment Plant Chemical Hazards report.

17. Properties and Concerns of Cerium (IV) Nitrate: This chemical arrives as a liquid and is extremely corrosive to stainless steels (it chemically machines the surfaces at rates approaching several mils per hour). It is also a strong oxidizer of organic materials. It is not clear if the Contractor has considered all of these properties, particularly the impact of residual concentrations of cerium (IV) as it is transferred through stainless steel piping to waste management. Also, the means for introducing the cerium (IV) solution into the cell area is not clearly explained, and temporary connections may be necessary.

18. Properties and Concerns of Permanganate: It is not clear if the Contractor appreciates the properties of this chemical. In particular, all permanganate solutions decompose at slow but significant rates, and leave a fine coating of manganese dioxide that is difficult to remove. Permanganate solutions can violently react with organic materials and accelerate the aging and embrittlement of elastomers and seals (e.g., in pumps and seals).

19. Properties of Peroxide: The Contractor recognizes peroxide as a strong oxidizer. However, explosive catalytic decomposition can occur if typical peroxide concentrations (30-50 vol%) are introduced to surfaces or solutions containing transition metals (e.g., rust, iron, nickel, manganese). Industrial experience has shown this to be capable of rupturing pipes. Residual concentrations will also attack and embrittle elastomers and seals (e.g., in pump areas). The means for introducing the peroxide solution into the cell area is not clearly explained, and temporary connections may be necessary.

20. Carbon Dioxide: The proposed facility would use large quantities of carbon dioxide (for surface decontamination), and has a relatively large tank containing liquid CO2. However, potential incidents involving this material, such as asphyxiation, do not appear to have been seriously considered in the analyses. It would seem that CO2 or breathing air monitors of some type might be needed and could be identified as safety controls.

21. Concerns about Silica & Zircon Sand: There are several inconsistencies with regards to the "Acute Toxic" and "Chronic Toxic" data for Silica and Zircon Sand, as shown in Tables 4 and 8 of the document.

    Table 4

    a) Silica acute and chronic toxic data needs to be the same as that of zircon sand as presented in Table 8. The rationale is that it is a known significant health hazard per NIOSH and Occupational Safety and Health Administration (OSHA).

    b) Zircon sand acute and chronic toxic data is listed as NA in Table 4. This needs to be replaced with the data presented in Table 8 for zircon sand.

    Table 8

    a) Silica acute toxic data needs to read same as that for zircon sand.

22. Concerns about a 20,000 Gallon Leak of 12.2 M Nitric Acid or 19 M Sodium Hydroxide: Table 1 of the Waste Treatment Plant Chemical Hazards report states that there will be 20,000 gallons each of concentrated nitric acid (12.2 M) and sodium hydroxide (19 M) on site. Given that both chemicals are hazardous and very reactive, the report needs to discuss in detail: (I) the rationale for using these very large onsite inventories in the proposed BNFL Inc. design, and (II) the emergency plans for a 20,000 gallon leak of either of these chemicals due to a leak in the discharge piping for these chemicals. Prudent design in accordance with AIChE guidance (essentially a chemical version of the ALARA principle) would dictate that provisions should be made to promptly empty the contents of the tank into temporary holding vessels and, as far as possible, the piping should be designed to eliminate the possibility of a large leak, for example, by using a top entering

tank discharge route and eliminating the bottom discharge from the tank as depicted in the Figure in Appendix A. Other process industry standards (AIChE, American Petroleum Institute, ANSI) would likely require vacuum and over-pressure protection, venting, NOx abatement on the vent(s), and management measures (coding and inspections). These standards would likely apply to other tanks in the facility as well.

23. Ambiguities in the Nitric Acid Scenario: There are inconsistencies and ambiguities. For modeling the nitric acid spill, a 5,000 gallon quantity of 12.2 M nitric acid is used. However, Table 1 cites up to 20,000 gallons present onsite and only one tank is identified. The basis for the selection of the worst case temperature of 95°F and meteorological conditions for dispersion of 1m/s is not stated. Other points of needed clarification include the EPICODE input data and other model parameters selected for the equation on page 43, including the partial vapor pressure of nitric acid at a weight percent of 61.1 percent and at 95°F. Per the last sentence, paragraph 5, page 43 - "The partial pressure of nitric acid only must be used in the above equation, or the source term will be overestimated." It is not clear if this overestimation is stipulated in or caused by the use of EPICODE.

24. Model Selection, Validation, and Comparisons: No copy of EPICODE (commercially available) was supplied to verify calculations for modeling worst-case scenario. There was insufficient model input data provided in the report to confirm or compare the EPICODE model results using a similar model, ARCHIE, which is a program initially developed in 1989 and available on the EPA public domain website. This program uses hydrazine instead of water as the reference liquid. It would be beneficial to include the rationale for using the EPICODE model instead of or in addition to one of the other EPA chemical release programs such as ALOHA, SLAB, ARCHIE or TSCREEN for various release scenarios. For example, EPICODE is a modeling and simulation emergency release program acceptable to DOE and EPA; ARCHIE also considers fire and explosion hazards, which are not fully presented in this report. It is not clear that the Gaussian dispersion formulas used to calculate the dispersed volatilized chemical concentration downwind from the spill are a better representation for this particular site and chemical (nitric acid). Some model references reviewed include:

   a. ALOHA and ARCHIE: A Comparison, Mary Evans, Report No. HAZMAT 93-2, April 1993, Modeling and Simulation Studies Branch, Hazardous Materials Response and Assessment Division, Office of Ocean Resources Conservation and Assessment, National Oceanic and Atmospheric Administration, Seattle, Washington 98115.
   b. ARCHIE Model, U.S. EPA Region 7 Website.
   c. Atmospheric Dispersion Modeling Resources, Second Edition, Emergency Management Advisory Committee, Subcommittee on Consequence Assessment and Protective Action, U.S. DOE. March 1995.

25. Rationale for Only Analyzing Nitric Acid Spills: Nitric acid is the only chemical modeled in an emergency release scenario. It would seem that other chemicals, such as sodium hydroxide and ammonia, might also pose concerns. It would be helpful to have the basis stated for concluding that 12.2 M nitric acid was determined to be the only chemical with appreciable volatility.

26. Heat Tracing of Nitric Acid Equipment and Lines: All the tanks/associated piping in the Process Description section (3.1.1 through 3.1.5), except the 12.2 M nitric acid tank, have heat tracing and insulation for freeze protection. Some basis would be anticipated to support this.

27. Nitric Acid Coaxial Piping - (Ref: 3.1.1, page 8): The report states that "piping outside the berm will be coaxial piping" to protect personnel from being exposed to nitric acid. It is advisable to look at the pressure side of acid piping inside the berm to ensure that no personnel will be exposed to the acid accidentally. Sometimes the piping inside the berm may be close enough to personnel outside the berm to present a potential hazard in the event of an acid leak.

28. Completeness of Interaction Matrix: The matrix of possible interactions of between different chemicals (page 29) does not include the interactions between utilities (steam and water, especially steam) and chemicals. Steam is used to provide heat to the urea unit (3.3.1, page 24) and the Cs HNO_3 recovery system (pages 33-34). It may not be a bad idea to consider the possible interactions between steam and the chemicals. For the sake of completeness, the interaction matrix may be expanded to include chemicals, materials of construction, utilities, operator(s), energy source, and others (air/water/land).

29. Stack Release Concentrations: It would seem that the postulated NOx release coming out of the stack at approximately 2000-3000 ppm exceeds the 100 ppm IDLH at the stack and quite possibly at the ground. Not knowing the height of the stack, flow rate, etc., of the system, it would not be reasonable to give a rough order estimate on the ground level NOx. However, BNFL Inc. definitely needs to address this issue by modeling, surrogate testing, and also possibly looking at other similar facilities to determine the ground level concentrations and emergency response procedures. Simply cutting off the feed does not seem to be an adequate response or control strategy.

30. Pneumatic Transfer Systems for Glass Former Powders: The Contractor intends to use a dense-phase pneumatic transfer system for the transport of the glass former powders. Due to the higher operating pressures involved (typically around 50 psig), a leak in the piping, such as at a coupling or a diverter valve, can result in an intense spray of dust into the immediate environment. Furthermore, the dust from some of the powders is hazardous. For example, silica (a major ingredient of the glass formers) is carcinogenic and has an OSHA Permissable Exposure Levels (PEL) (respirable fraction) and American Council of Government and Industrial Hygienists (ACGIH) threshold limit value (respirable fraction) of 100 micrograms/m^3 (this is less than 0.1 ppm). Consequently, it would be anticipated that a leak scenario involving this system would be explored to ensure that worker safety is not compromised.

31. Other Pneumatic Transfer System Features: There are other design safety considerations for these systems, including the following:

    a. Pipe couplings and hangers need to be designed to withstand pressure surges of two to three times the normal operating pressure. Such surges are normally encountered in dense-phase pneumatic transport systems and can induce significant, hammer-like stresses.

b. Pneumatic transport systems are prolific generators of static electricity and the design must incorporate measures that prevent or mitigate potential accidents initiated by static charges.

c. There are likely to be ASME code requirements and stamps for many of the components in the system (e.g., vessels).

32. Multiple Uses for Some Components: Some lines and components will be used to mix and/or feed different chemicals at different times. It is not clear if an adequate analysis has been performed to address the potential, inadvertent mixing of incompatible chemicals in these lines and components. For example, on different occasions, the MCCMT (Miscellaneous Cold Chemicals Mix Tank) will be used for the preparation and addition of sugar shimming solutions and decontamination chemicals (e.g., usually nitric acid based in TWRS-P - see Section 3.3.3 of the report). Since the preparation of chemicals in the MCCMT will be a manually controlled operation subject to human factors in addition to component failures, the Contractor's design may be vulnerable to the inadvertent generation of copious quantities of NOx from the unplanned mixing of these chemicals. It would be anticipated that such scenarios would be analyzed and the appropriate controls identified. As an aside, this was the reason that the addition of nitric acid to the shim tank was restricted at West Valley Demonstration Project even though the tank was vented to a venturi scrubber.

33. Diesel Fuel Concerns: Large quantities of diesel fuel will be stored and handled at the facility. It would be anticipated that there would be a discussion of the safe handling of this fuel.

3.0 AREAS OF REVIEW
3.7 FIRE PROTECTION

3.7.1 STANDARDS APPROVAL PACKAGE (SAFETY REQUIREMENTS DOCUMENT AND HAZARDS ANALYSIS REPORT) AND INITIAL SAFETY ANALYSIS REPORT

Although there were fire protection comments generated during the Standards Approval Package (SAP) and the Initial Safety Analysis Report (ISAR) reviews, most of these comments were concerned with clarifications or omission of details. The design had not yet progressed to a point were an actual description of fire protection systems could be developed and reviewed against our existing review criteria.

From these reviews, comments and questions were raised concerning:

1. Discounting of fire in the hazard analysis due to non-conservative assumptions regarding combustible loading.

2. Failure to reference specific applicable subsections in National Fire Protection Association (NFPA) Codes in the "tailoring process."

3. Need to determine a minimum fire resistance for fire barriers.

4. Need to designate building construction type as per NFPA 220.

5. Manual fire fighting requirements and coordination with the Hanford Fire Department.

6. Need to properly consider heating, ventilation, and air conditioning (HVAC) filter fire in hazards analysis.

7. Need to commit to specific NFPA standards.

The actual comments may be found in the comment transmittal letters[40].

Most of these concerns were resolved through the upcoming Topical and Level 1 Meetings on fire protection and the documentation provided to support those meetings.

3.7.2 TOPICAL MEETING ON FIRE PROTECTION (FEBRUARY 1999)

The first detailed information provided by BNFL on fire protection was provided in the "Transmittal of Information for the February Topical Meeting[41]." This letter was written as a response to U.S. Department of Energy/Regulatory Unit (DOE/RU) comments in their Initial

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[40] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, October 17, 1997.
Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, February 6, 1998.

[41] Edwards letter to D.C. Gibbs, U.S. Department of Energy, February 09, 2000.

Safety Evaluation Report and described elements of the Tank Waste Remediation System-Privatization (TWRS-P) fire safety program in more detail than was provided in the ISAR and SAP submittals. In this submittal information was provided regarding:

1. A description of the TWRS-P fire protection program.
2. TWRS-P Facility Design and Layout Configuration.
3. Life Safety Features.
4. Fire Protection System.
5. Fire Protection Personnel.
6. Fire Hazards Analysis - Methodology and Assumptions.
7. Analysis to Achieve a Safe State.
8. Seismic/Fire Interactions.
9. Applicability of Codes and Standards.
10. Fire Resistance of Fire Area Boundaries.
11. Structural Steel Fireproofing.
12. Fire Doors, Ducts, Dampers, and Penetration Seals.
13. Employment of Automatic Suppression Systems.
14. Filter Plenum Fire Protection.
15. Compliance with Standards for Flammable Gases.
16. Fire Safety Features for Mechanical/Electrical Systems Areas.
17. Baseline Needs Assessment.

The information provided closed many of the concerns of both NRC and the RU. Concerns that remained unresolved or came about as result of the submittal were:

1. Interfacing with the Hanford Fire Department and the Need for an Onsite Fire Brigade.
2. Extent of Fireproofing of Structural Steel.
3. Extent of Employment of Sprinkler Systems.
4. Filter Plenum Protection.

A revised version of the February 9 submittal was submitted to the RU on March 4, 1999. This version clarified some of the responses to address RU concerns, however, it did not resolve the NRC concerns listed above.

3.7.3 LEVEL 1 MEETING ON FIRE PROTECTION (SEPTEMBER 1999)

The next meeting on fire protection involving the NRC staff was the first Level 1 Meeting held on September 29, 1999 at BNFL Inc. in Richland, Washington.

In regard to interfacing with the Hanford Fire Department, no definitive decision had been made as yet. As presently planned, the TWRS-P project will use a baseline needs assessment based on the results of the Fire Hazards Analysis (FHA) which will be reconciled with the capabilities of the Hanford Fire Department (HFD). A question arose about training of facility staff in use of fire extinguishers. BNFL Inc. indicated that only selected operations people would be trained and that there was no plan at present to rely on facility personnel for fighting incipient fires.

The meeting minutes specified that fire-exposed structural steel that supports a fire barrier (wall, floor, ceiling) will be fireproofed to maintain the fire-resistance integrity of the building. BNFL inc. also stated that their approach would be consistent with NFPA 220, the NRC criteria, although it might not be consistent with the Uniform Building Codes (UBC) (the DOE criteria). During the meeting, however, the possibility of not fire-proofing structural steel in areas of low combustible loading was also discussed. This type of approach would not meet either DOE or NRC criteria. A letter[42] providing comments on the Explosive Hazards I and II topical meetings also contained a short description of the NRC criteria for building construction and expressed concern about the stated BNFL Inc position regarding fire proofing of structural steel.

The use of sprinklers was discussed at length between the RU and BNFL Inc. Most of the discussion was centered around the desire of the RU (and NRC) for BNFL Inc. to establish a definitive set of criteria for use of sprinklers in a given area. BNFL Inc. replied that they would rely on the FHA and judgement because of the uniqueness of every different fire area in terms of type and distribution of combustibles, heat release rates, ventilation, etc. Some tentative guidance that BNFL Inc. provided included the following consensus reached by the project fire protection engineers:

1. C5 Areas: High potential for contamination areas having low fireloading. It was the understanding of the RU and NRC that BNFL Inc. did not want to provide sprinklers or steel fire-proofing in some C5 areas.

2. C2 Areas: Low potential for contamination areas having anticipated in-situ and transient combustibles. These areas would likely be sprinklered.

3. C3 Areas: Moderate contamination areas with varying degrees of combustible loading and safety significance. Sprinklers are expected to be used selectively in these areas with justifications provided where sprinklers are not used.

This guidance, in the opinion of the RU and the NRC staff, appeared to leave too much uncertainty to alleviate the concern about sprinkler coverage.

Water spray protection for filter plenums was not discussed in detail during the meeting. It was mentioned that the high efficiency particulate air (HEPA) filters would be canister type rather than the U.S. rectangular-type filters, and in the opinion of BNFL Inc., not as likely to burn. Also, there was discussion of providing automatic water sprays with an optical detector to signal the control valve. BNFL Inc. said that the issue was still being considered and did not wish to make a commitment.

BNFL Inc. also used the meeting to back-off somewhat on their earlier commitment to provide a seismic supply and standpipe system. BNFL is considering the use of a mobile source such as a tanker truck and pumper. The RU noted that the Hanford Fire Department tanker trucks are used primarily for brush fires and are not kept filled in the winter. NRC staff expect resolution of this issue without significant interaction with BNFL Inc.

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[42] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, November 4, 1999.

3.7.4 LEVEL 1 MEETING ON FIRE PROTECTION (MARCH 2000)

As of the last Level 1 Meeting in fire protection in March 2000, there were three issues in fire protection of primary interest to the RU and NRC staffs. These issues are structural classification of the process buildings, structural steel fire proofing, and protection of the final HEPA filters from the effects of fires. The RU also identified confinement system integrity as a fire protection issue. However, since this issue involves basic design of the ventilation system, NRC staff would prefer to make this issue a plant systems/radiation protection issue for the time being.

In regard to the process building construction type issue, BNFL proposes a construction type which would not meet the NRC requirements for Type I construction as per NFPA 220 nor the fire resistive construction types required by a strict interpretation of the UBC. The RU has commented to BNFL Inc.[43] that a justification for the selection of a nonfire resistive construction type would have to include identification and analysis of building areas with unusual or concentrated hazards, bounding fuel packages, and critical structural members whose failure due to fires could structurally undermine a significant portion of the facility. The analysis would also be expected to address any fire protection administrative controls, such as transient combustibles and vehicle access to the proximity of process buildings (e.g., control of exterior fire exposure hazards), which are relied on. The NRC stated[44] to the RU that the staff was in agreement with the RU criteria for resolution of this concern.

The steel fire proofing issue is related to the construction type issue. I f fire resistance of outside walls is not required, fireproofing of structural steel for these walls would not be required. In addition, BNFL Inc. wishes to avoid fireproofing of structural steel in areas of low combustible loading. The RU has commented to BNFL (Footnote 43) that the RU continues to expect that TWRS-P structural fire proofing will meet the requirement of the Uniform Building Code. The RU also expects that any type of equivalency analysis will demonstrate that there are no fire hazards that present a potential threat to structural steel integrity. The NRC stated (Footnote 44) to the RU that NRC staff was in agreement with their comments and added two additional points:

1. Lack of steel protection based on intended use would severely limit any future modifications for use of that area, and

2. Lack of fire suppression capability along with lack of steel protection in the same area makes administrative control of combustibles the only defense measure and makes the fire protection in the area a potential vulnerability that would be identified in a Fire Hazards Analysis (FHA)/Integrated Safety Analysis (ISA).

BNFL Inc. stated in the fire protection level I meeting that no water sprays would employed in filter plenums based on the superior qualifications of the circular filters to be used. Both NRC and DOE fire protection criteria call for water spray protection in the final filter bank of the

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[43] Gibbs, D.C., U.S. Department of Energy, letter to M. Bullock, BNFL Inc., May 03, 2000.

[44] Tokar, M., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, June 05, 2000.

confinement system. At this time the RU (May 03, 2000 - Footnote 43) is requesting clarification on the BNFL Inc. position.

3.7.5 PRESENT STATUS WITH REFERENCE TO POINT PAPER

Other fire protection issues that appear to be moving toward resolution but will require more detailed information are listed below. Some, but not all, of these issues are discussed in the TWRS-P Fire Safety Point Paper[45].

1. Emergency lighting - Emergency lighting was addressed in the Part b-1 deliverable and described as "not important to safety." No engineering evaluations were provided to address safety significance and operating requirements for emergency lighting.

2. Fire detection and alarms - Fire detection and alarms were addressed in the Part b-1 deliverable and it was stated that they would be designed in accordance with NFPA 72. Universal coverage appears to be provided but information on the types of detectors to be placed in various areas was not provided.

3. Fire water pump design - A conceptual drawing of fire pump placement and plumbing was provided in the Part b-1 deliverable. Commitments were made to follow NFPA 20 and 24. Fire suppression demand calculation and details of pump plumbing will need to be provided.

4. Seismic water supply - No details provided as yet. A tanker truck and pumper have been initially discussed.

5. Baseline needs for manual fire fighting and coordination with Hanford Fire Department: A baseline needs assessment will be required for the Construction Authorization Request (CAR) but will need more of the plant design to be completed. Discussions have taken place with the Hanford Fire Department but a formal interface is awaiting contract resolution between DOE and the project.

Based on discussion with BNFL Inc. all of these issues were expected to be resolved at the time of the CAR submittal.

Other potential issues that were discussed in the Fire Safety Point Paper but appear to be resolved include:

1. Fire area boundaries.
2. Treatment of combustibles in FHA.
3. Spurious actuations and circuit protection.
4. Penetration seals.

In the Issues Close Out Topical Meeting of June 27, 2000, BNFL Inc. identified three issues in fire protection as being open issues:

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[45] Leach, M.N., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, March 02, 2000.

1. Use of Automatic Sprinklers.
2. Structural Steel Fireproofing.
3. Hanford Fire Department Interface.

BNFL Inc. claimed closure in the area of automatic sprinklers because of the basic consensus between the project and RU described in the meeting minutes from the March Level 1 Meeting; that is, for areas other than C5 areas, the project may omit sprinklers only with FHA justification and RU approval and a preliminary FHA will be performed on selected parts of the Pretreatment Building to demonstrate the projects philosophy on sprinkler coverage. The NRC agrees that based on the BNFL Inc. commitments, this issue may be considered closed.

In regard to structural fireproofing, the project still believes that verbatim compliance with the UBC (and due to similar requirements, the NRC standard review plan) is not warranted by the hazards. The architects are preparing an equivalency evaluation for review by the RU. Based on this, the NRC would not consider the issue closed until the equivalency proposal has been reviewed and approved.

Although technical interface issues were discussed with the Hanford Fire Marshal on May 10, 2000, the interface is on hold pending contract resolution between DOE and the project. The NRC would not consider this issue closed until a contract has been completed between the HFD and the project.

The NRC staff considers water sprays in filter plenums to still be an open issue although this was not addressed by BNFL Inc.

3.0 AREAS OF REVIEW

3.8 EXPLOSION PROTECTION ISSUES

3.8.1 STANDARDS APPROVAL PACKAGE (SAFETY REQUIREMENTS DOCUMENT AND HAZARDS ANALYSIS REPORT) AND INITIAL SAFETY ANALYSIS REPORT

Although there were numerous fire protection comments generated during the Standards approval Package (SAP) and the Initial Safety Analysis Report (ISAR) reviews, most of these comments were concerned with clarifications or omission of details. The design had not yet progressed to a point were an actual description of explosion control systems could be developed and reviewed against our existing review criteria.

From these reviews comments and questions were raised concerning:

1. Flammable Gases in waste receipt tanks.
2. Presence of ammonium nitrates and organic vapors in melter effluents.
3. Buildup of ammonium nitrate in offgas system ducts and filters.
4. Capacity and reliability of passive hydrogen venting system.
5. Monitoring of hydrogen and other flammable gases.

The actual comments may be found in the comment transmittal letters (Footnote 40).

3.8.2 DESIGN SAFETY FEATURES TOPICAL MEETING—HYDROGEN CONTROL (JANUARY 1999)

The first relatively comprehensive treatment of the hydrogen control problem was presented at the Topical Meeting on hydrogen control as the first example of design safety features. BNFL Inc. presented calculations of hydrogen generation rates that the U.S. Nuclear Regulatory Commission (NRC) staff questioned as possibly unconservative[46,47]. The proposed means of keeping the hydrogen within acceptable limits if active ventilation should fail was a passive ventilation system relying on the density differences between air and hydrogen as well as the thermal gradient between the vessel and the cell. Hydrogen monitoring was recommended for only a short time during initial plant operations. This approach was also questioned in the February and March 1999 comment letters. The NRC staff directed its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA), to provide an independent calculation of hydrogen generation rates and indicate areas of significant uncertainty. CNWRA responded with the report "Review of BNFL Inc. Design Safety Features Deliverable: Hydrogen Control in High Level Waste Storage Tanks[48]." In this report, CNWRA showed that a rate of hydrogen

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[46] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "Review of Hydrogen and Explosion Information Presented at the Topical Meeting on January 29, 1999." February 12, 1999.

[47] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "Comments Concerning BNFL Inc.'s Design Safety Features Submital." March 18, 1999.

[48] Center for Nuclear Waste Regulatory Analyses (CNWRA). CNWRA 99-001, Rev 1, "Review of BNFL Inc. Design Safety Features Deliverable: Hydrogen Control in High Level Waste Storage Tanks." CNWRA: San Antonio, Texas. April 1999.

generation over three times that calculated by BNFL Inc. could be calculated using equally valid inventory assumptions as those used by BNFL Inc. Even greater rates could be determined by taking into account uncertainty in parameters and off normal process conditions. Because the successful operation of the passive ventilation system was sensitive to hydrogen generation rates, the results of the CNWRA studies raised considerable concern in the NRC staff and the RU about the reliability and adequacy of the passive ventilation system.

3.8.3 EXPLOSIVE HAZARDS TOPICAL MEETING I (AUGUST 1999)

BNFL Inc. reevaluated the explosion hazard due to radiolytic hydrogen generation as well as other possible sources and prepared reports for two topical meetings, Explosive Hazards I and II, which were held in August and September 1999, respectively. The August Topical Meeting (Explosive Hazards I) addressed hazards from melter steam explosions, explosions from nitrate organic reactions in the melter offgas system, over pressurization of an ion-exchange column, ammonium nitrate explosion in the offgas system, and a sugar dust explosion in the feed preparation vessel. Although none of these events were considered credible by BNFL Inc., all of these event were left as potential issues by NRC and the U.S. Department of Energy/Regulatory Unit (DOE/RU) pending further information and analysis[49]. A resubmittal of the August 1999 topical report in March 2000 and information from design review meetings alleviated some of the NRC concerns about steam explosions, nitrate-organic reactions, and sugar dust explosions[50]. NRC staff present concerns about steam explosions are primarily the potential for a refractory failure allowing molten glass to contact water in the cooling jacket. Potential explosions caused by the contact of water and the cold cap were addressed by the BNFL Inc. responses to RU comments. Also, the BNFL Inc. process does not appear to be as vulnerable to radiological releases from organic-nitrate and sugar explosions as originally determined from the first submittal, although some questions still remain. Potential explosions from ammonium nitrate formation and over pressurization of the ion exchange column are still considered as open issues.

3.8.4 EXPLOSIVE HAZARDS TOPICAL MEETING II (SEPTEMBER 1999)

The September Topical Meeting (Explosive Hazards II) was concerned with the potential for explosion of hydrogen gas in approximately 40 process tanks. At this time an active ventilation system for hydrogen build-up was proposed to take the place of the passive release system that was proposed earlier, probably for a single tank. This active system consisted of an air extract system with two 100 percent fans and one 100 percent backup fan. Process air is used for dilution during normal operation. An air vent system is provided for loss of offsite power (with loss of process air) conditions. Although the staff considered this design a significant improvement over the passive system, there were concerns from the presentation and report (Footnote 49). These concerns included:

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49 Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "NRC and CNWRA Comments on the Explosive Hazards Topical Meetings I and II." November 4, 1999.

50 Tokar, M., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "NRC Response to March 23, 2000 Letter Concerning the Resubmittal of the August and September 1999 Topical Meeting Reports on Explosive Hazards." April 27, 2000.

1. Hazard evaluations based on normal operating conditions rather than off normal.
2. Rupture of vessel vent jumper was not considered in hazard evaluation.
3. Loss of ability to maintain pressure control in vessel vent system was not considered.
4. Large surges in offgas were not considered.
5. The worse case detonation scenario was not properly evaluated.

A review of the resubmittal in February 2000 also led to a re-evaluation of the open issues. The NRC staff concluded that rupture of the vessel vent jumper and loss of ability to maintain pressure control can probably be nearly eliminated by an expanded set of performance requirements on the system. The other concerns remain however.

3.8.5 NRC ASSESSMENT OF PRESENT STATUS

In regard to radiolytic hydrogen explosions, the point paper (Footnote 45) accurately reflects the status of the hydrogen control system. An issue that was not resolved at the explosive hazard topical meetings was the need for monitoring. This issue was addressed at the June 2000 Topical Meeting on Close Out of Open Issues. BNFL Inc. has argued that air flow monitoring meets the requirement in Section 3-4.1 of the National Fire Protection Association (NFPA) 69, "Instrumentation shall be provided to monitor the control of the concentration of combustible components." The NRC staff agrees with this argument. BNFL Inc. also argued that the concentration of hydrogen would always be below the measurement threshold of normally used monitoring instruments except in the case of system failure; which BNFL Inc. has calculated to be incredible. Unfortunately, BNFL Inc. did not provide any calculations to show what the range and probability distribution of hydrogen concentrations would be under normal operating and expected off normal conditions could be. When the project provides these calculations and the RU accepts the reliability of the active venting system, this issue should be closed.

The point paper also discusses sugar dust as another significant potential source of an explosion. However, the addition of inerting agents to granular sugar before it's pneumatically conveyed to the feed hopper and the use of gravity feed to the melter feed preparation vessel are expected to preclude likely explosion scenarios. The project has not yet addressed the selection of codes and fire protection measures nor has it evaluated the potential for and consequences of a vehicle accident in the sugar storage area (Footnote 50).

Ammonium nitrate formation and organic vapors were addressed briefly in the point paper under fire protection for filter plenums. The technology as presented by BNFL Inc. appears capable of preventing explosive buildups in the offgas system and this should probably be the strategy as opposed to using fire protection methods as discussed in the point paper.

Steam explosions in the melter and over pressure in the ion exchange column were not considered to be fire protection issues and were not addressed in the point paper.

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3.0 AREAS OF REVIEW
3.9 ENVIRONMENTAL PROTECTION

3.9.1 ENVIRONMENTAL IMPACT STATEMENT

The U.S. Department of Energy (DOE) has published an environmental impact statement (EIS) on Tank Waste Remediation Systems (TWRS)[51]. This EIS identified as the preferred alternative the phased implementation approach (Phase I/II), with separations and treatment in external facilities. Vitrification is identified for immobilizing both the low activity waste (LAW) and high level waste (HLW). The EIS analyzed impacts and accident scenarios for the proposed facilities with the intent of enveloping any specific designs and process approaches that would be developed by the TWRS-Privatization (TWRS-P) program. Scenarios involving spray leaks were identified as having the potential for the most significant effects. The NRC staff did not review the EIS as part of its involvement.

3.9.2 STANDARDS APPROVAL PACKAGE (SAFETY REQUIREMENTS DOCUMENT AND HAZARDS ANALYSIS REPORT) AND INITIAL SAFETY ANALYSIS REPORT

The standards criteria in Section 5.3, "Environmental Radiation Protection," of Volume II of the Safety Requirements Document (SRD) outlines the Contractor's approach for the environmental protection (EP) program. Although the safety criteria (SC) defines the limits of the EP program, this material does not appear to be well supported throughout the remainder of the SRD package (Reference 3). For example, environmental monitoring is committed to (SRD criteria 9.5-1 and 9.5-2) but not described in relation to the anticipated plant operations. In addition, the SRD and Hazards Analysis Report (HAR) inconsistently commit to protection of the environment and frequently limit EP in terms of the worker and the public. Other major comments included:

1. Consistency and integration of environmental submittals with respect to other submittals and with the regulatory agencies (DOE/Regulatory Unit (RU), DOE Richland Operations Office (RL), DOE Office of Environment, Safety and Health (EH), U.S. Environmental Protection Agency (EPA), and the State of Washington).

2. Application of safety categorizations for EP (i.e., structures, systems, and components (SSCs) that prevent or minimize environmental contamination).

3. Assessment of incidents for discharges to the vadose zone.

4. No clear identification of effluent and emission points on the site plan and facility information.

5. Description of offgas emissions and controls, including the effect from the emergency offgas system and iodine-129 removal systems.

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[51] Department of Energy (U.S.)(DOE). DOE/EIS-0189, "Tank Waste Remediation System, Hanford Site, Richland, Washington, Final Environmental Impact Statement." DOE: Richland, Washington. August 30, 1996.

The Initial Safety Analysis Report (ISAR) provided little additional detail on the EP approach. It does refer to a separate "Environmental Report" by the Contractor that presents a broad overview of monitoring and mitigation approaches. The U.S. Nuclear Regulatory Commission (NRC) comments (see Footnote 19 on page 40) noted that detail is not provided on such items as media to be sampled, sampling locations, sampling frequencies, methodologies, quality assurance/quality control (QA/QC), action levels for elevated measurements or other components that might be expected in an EP program (e.g., see the standard review plan - Reference 15). Furthermore, the ISAR focused on impacts to the workers, the co-located workers, and the public, and did not include an assessment of the effects upon the environment (e.g., from accidents). Thus, it was not clear if adequate protection of the environment is achieved by SSCs that protect the health and safety of the public (Reference 5 —the ISAR—Section 3.3.6, page 3-35).

Similar NRC comments were submitted later[52] requesting information on how environmental protection was supported by the implementing codes and standards listed in the (SRD) and updated Integrated Safety Management Plan (ISMP).

3.9.3 DESIGN SAFETY FEATURES

The Design Safety Features (DSF) submittal did not discuss EP in any significant detail.

3.9.4 RELATED ACTIVITIES AND TOPICAL MEETINGS

The Contractor was working with the EPA Regional Office, Washington Department of Ecology, and Washington Department of Health on environmental topics and permitting issues . The work required for permit applications started in 1998. Test results and modeling efforts were required for such essential items as melter offgas and vessel vent streams.

Additional NRC comments on Contractor responses were submitted to the DOE/RU[53]. Many of the comments dealt with compliance with the Washington Administrative Code (WAC), and there was an initial view that compliance with the WAC was limited to air discharges. The facility design was such that no radioactive liquid discharges were to be made because all radioactive liquids were to be routed through the existing Effluent Treatment Facility.

A BNFL Inc. letter forwarded standards identified for safety criteria[54]. A second set of NRC comments on the Contractor Environmental Radiological Protection Program (ERPP) and its

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52 Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "NRC TWRS Section Comments on BNFL's Revised SRD and ISMP." July 23, 1998.

53 Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "U.S. Nuclear Regulatory Commission's Comments on BNFL's Tank Waste Remediation Systems Safety Criteria for Environmental Radiological Protection Program and for Environmental Radiological Monitoring." December 31, 1998.

54 Edwards, BNFL Inc. letter to D.C. Gibbs, Department of Energy, RE ERPP. January 18, 1999

revision were submitted to the DOE/RU[55]. The DOE/RU disposition of the comments were forwarded to NRC on March 17, 1999.[56] NRC reviewers felt that as low as reasonably achievable (ALARA) criteria from the WAC was the correct citation, since 10 CFR Part 835 refers to ALARA only in the context of an occupational radiation protection program. In addition, there are no environmental requirements in 10 CFR Part 835. However, DOE/RU felt that 10 CFR Part 835 was adequate. The correct citation for Best Available Radionuclide Control Technology (BARCT) for air emissions is WAC 246.247.120, Appendix B.

For DOE sites, the major Federal law regulating air emissions is the Clean Air Act (CAA). The requirements for obtaining permits is to ensure that any emissions of listed hazardous air pollutants (HAPs), including radionuclides, comply with the National Emissions Standards for Hazardous Air Pollutants (NESHAPS), Section 112. The emission limits are given in State Implementation Plans (SIPs). NRC reviewers pointed out that even though DOE intended to use the existing monitoring network to collect air samples during construction and operation, the Contractor needed to conduct environmental surveillance to demonstrate compliance with the ERPP.

Other clarifications were required regarding the concepts of "groundwater protection" versus "groundwater monitoring." The TWRS-P facility was to be operated under the Resource Conservative Recovery Act of 1976 (RCRA) as a Treatment Storage and Disposal (TSD) facility, and groundwater monitoring was to be addressed as part of RCRA permit requirements. NRC reviewers felt that although International Organization for Standardization (ISO) 14001, which was selected by the Contractor for groundwater monitoring, provided a sound management plan framework, specific codes and standards would need to be identified.

NRC reviewers pointed out that sections of the WAC related to solid/hazardous wastes should have been included in the regulatory basis, since the state of Washington had regulatory authority under RCRA. The TWRS-P facility was to be added to an existing site-wide RCRA permit.

Other issues associated with environmental radiation protection include such topics as mixed waste (which involves dual regulation), pollution prevention, waste minimization, and contamination control.

DOE/RU responses and NRC comments can be grouped into the following categories: (1) compliance demonstration, operating procedures, and administrative procedures; and (2) American National Standards Institute (ANSI)/ISO-14001 as an adequate implementing standard.

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[55] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "U.S. Nuclear Regulatory Commission Comments on BNFL's Tank Waste Remediation Systems Safety Criteria for Environmental Radiological Protection Program and for Environmental Radiological Monitoring." February 11, 1999.

[56] Gibbs, D.C., U.S. Department of Energy, letter to R.C. Pierson, U.S. Nuclear Regulatory Commission, March 17, 1999.

The TWRS-P Contractor had started submitting permit applications at the time of the termination. These were generally based upon the design as it existed in late calendar year 2000. Significantly, the approach intends to store failed melters (i.e., melters removed from service due to refractory age, etc., but not necessarily failed as a breach) during the operational phase of the facility, followed by near-surface disposal during the decommissioning period. The NRC staff did not review the specific details of this approach. However, based upon meetings in the Spring of 2000, the approach relies upon movement of intact melters in overpacks, with total weights of several hundred tonnes. Some vitrified waste materials would remain within the failed melters. For the failed HLW melter, the great majority of the vitrified waste would have to be removed in order to meet near-surface disposal requirements. It was not clear how this could be accomplished for a melter without bottom drains.

3.9.5 NRC ASSESSMENT OF THE PRESENT STATUS

The NRC staff concludes that considerable effort and specificity remain to be done in this environmental area. This should be achievable concurrent with the advancement of the design

3.0 AREAS OF REVIEW
3.10 QUALITY ASSURANCE AND MANAGEMENT MEASURES

3.10.1 INTRODUCTION

The U.S. Department of Energy (DOE) has responsibility for the regulatory oversight of the Hanford Tank Waste Remediation System-Privatization (TWRS-P) contract. Should that responsibility be transferred to the U.S. Nuclear Regulatory Commission (NRC) at some future time, the responsibility for oversight of the TWRS-P quality assurance (QA) program for items relied on for safety (IROFS) would be part of the overall responsibility transferred. The application of QA to TWRS-P is a fairly complex subject, and the transfer of the QA regulatory oversight will also be complex. The complexities stem from the facts that:

1. DOE has regulations and guidance for QA that govern its facilities, and NRC has its own regulations and guidance for QA for NRC-licensed facilities.

2. The nuclear industry (represented by the American Nuclear Society) standard for QA, American Society of Mechanical Engineers (ASME) NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications,"[57] undergoes continual change.

3. NRC's regulations and guidance for QA and safety management measures for facilities licensed under 10 CFR Part 70 are currently being revised to address IROFS and a safety analysis.

This chapter discusses the DOE and NRC requirements for QA for TWRS-P and clarifies the status of the regulatory approach that has been taken.

3.10.2 DOE QA REQUIREMENTS AND GUIDANCE

The QA of DOE nuclear facilities such as TWRS-P is currently regulated by the DOE QA rule[58]. By letter of September 8, 1998 (D. Clark Gibbs to Maurice J. Bullock), DOE informed BNFL Inc. (the DOE TWRS-P contractor) that: "The Contractor's processes necessary to implement ISM (Integrated Safety Management) are . . . subject to, and the RU (DOE's Regulatory Unit) will evaluate them against, the requirements of the QA rule."

DOE has issued a generic guide[59] for implementing its QA rule that provides guidance as to what DOE considers an acceptable QA program. This guide includes many references. Before the references, the guide states that "The following references provide acceptable methods for implementing many of the requirements of 10 CFR 830.120.... No single reference fully meets

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[57] ASME NQA-1 is a consensus American National Standard issued every 3 years (with annual updates) by the American Society of Mechanical Engineers.

[58] Code of Federal Regulations, *Title 10, Energy,* Part 830, "Nuclear Safety Management," includes the U.S. Department of Energy Quality Assurance rule, (10 CFR 830.120, "Quality Assurance Requirements").

[59] Department of Energy (U.S.)(DOE). DOE G 830.120, Draft, "Implementation Guide for Use with 10 CFR Part 830.120 and DOE Order 5700.6C, Quality Assurance." September 1997.

all of the requirements. The principles, recommended approaches, and applications contained in these references may be used in conjunction with 10 CFR 830.120...to develop an effective management system to achieve quality."

One of the references listed in the generic DOE guide is ASME NQA-1-1994. ASME NQA-1-1994 has incorporated the ASME NQA Standard requirements of the previous editions of ASME NQA-1 and ASME NQA-2[60] into a single document: Part I of ASME NQA-1-1994 contains the prior ASME NQA-1 Standard requirements and Part II contains the prior ASME NQA-2 Standard requirements. Part III of ASME NQA-1-1994 contains the ASME NQA Standard guidance.

In addition, DOE has issued a specific guidance document for its QA reviewers to use when reviewing TWRS-P QA programs.[61] This guidance was based on the principles in the DOE QA rule and its generic guidance document. This TWRS-P specific guidance document was used by DOE and DOE contractor personnel when reviewing the BNFL Inc. QA program description.

3.10.3 NRC QA REGULATIONS AND GUIDANCE

Prior to this year, NRC regulations for domestic licensing of special nuclear material (SNM) require QA only for licensees who possess and use SNM in a plutonium processing and fuel fabrication plant.[62] However, a near-criticality incident at a low enriched uranium fuel fabrication facility prompted the NRC to review its safety regulations for applicants/licensees that possess and process large quantities of SNM.[63] As a result of this review the NRC staff recognized the need to revise its regulatory base for these licensees, particularly for organizations such as those that possess and process a critical mass of SNM. Therefore, the NRC has approved rulemaking[64] to amend 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." QA and management measures are addressed for wider application in the revision to 10 CFR Part 70.

The NRC QA regulatory requirements for domestic licensing of SNM apply to 10 CFR Part 70 SNM applicants/licensees authorized to possess and process a critical mass of SNM. Additional baseline design criteria (requirements) are included for new facilities such as the planned TWRS-P facility (if and when applicable) and for new processes at existing facilities.

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[60] American Society of Mechanical Engineers (ASME). NQA-2, "Quality Assurance Requirements for Nuclear Facility Applications, An American National Standard." ASME: 1994.

[61] Department of Energy (U.S)(DOE). RL/REG-96-01, Revision 0, "Guidance for Review of TWRS Privatization Contractor Initial Quality Assurance Program." DOE: Richland, Washington. October 1996.

[62] Code of Federal Regulations, *Title 10, Energy,* Part 70, "Domestic Licensing of Special Nuclear Material," (specifically 10 CFR 70.22, "Content of Applications," and its footnote reference to Appendix B of 10 CFR Part 50.)

[63] The results of this review are documented in: Nuclear Regulatory Commission (U.S.)(NRC). NUREG-1324, "Proposed Method for Regulating Major Materials Licensees." NRC: Washington, D.C. February 1992.

[64] Nuclear Regulatory Commission (U.S.)(NRC) . SECY-99-147, "Proposed Rulemaking—Revised Requirements for the Domestic Licensing of Special Nuclear Material." NRC: Washington, D.C. June 2, 1998. This is available on the web at http://www.nrc.gov/NRC/COMMISSION/SECYS/secy1999-147/1999-147scy.html.

The revised 10 CFR Part 70[65] indicates that QA is a part or subset of management measures as indicated in the 10 CFR 70.4 definition:

> Management measures means the functions performed by the licensee, generally on a continuing basis, that are applied to items relied on for safety, to ensure the itmes (sic) are available and reliable to perform their functions when needed. Management measures include configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, **and other quality assurance elements**. (Emphasis added.)

The following requirement regarding management measures in 10 CFR 70.62, "Safety Program and Integrated Safety Analysis," is from the revised 10 CFR Part 70:

> (d) *Management measures.* Each applicant or licensee shall establish management measures (including configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements) to provide continuing assurance of compliance with the performance requirements of §70.61. The measures applied to a particular engineered or administrative control or control system may be commensurate with the reduction of the risk attributable to that control or control system. The management measures shall ensure that engineered and administrative controls and control systems that are identified as items relied on for safety pursuant to §70.61(e) of this Part are designed, implemented, and maintained, as necessary, to ensure they are available and reliable to perform their function when needed, in the context of compliance with the performance requirements of §70.61 of this Part.

The baseline design criteria in the revised 10 CFR Part 70 address management measures in the following manner:

> §70.64 Requirements for new facilities or new processes at existing facilities.
> (a) *Baseline design criteria.* Each prospective applicant or licensee shall address the following baseline design criteria in the design of new facilities. Each existing licensee shall address the following baseline design criteria in the design of new processes at existing facilities that require a license amendment under §70.72. The baseline design criteria must be applied to the design of new facilities and new processes, but do not require retrofits to existing facilities or existing processes (e.g., those housing or adjacent to the new process); however, all facilities and processes must comply with the performance requirements in §70.61. Licensees shall maintain the application of these criteria unless the evaluation performed pursuant to paragraph (c) of this section demonstrates that a given item is not relied on for safety or does not require adherence to the specified criteria.
> (1) Quality standards and records. The design must be developed and implemented in accordance with management measures (including configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements) to provide adequate assurance that items relied on for safety will be available and reliable to perform

[65] Attachment 1 to the proposed rule noted in the previous footnote.

their function when needed. Appropriate records of these items must be maintained by or under the control of the licensee throughout the life of the facility.

The requirement that each applicant for a new 10 CFR Part 70 license must include a description of its management measures is specified in the following section of the latest draft of 10 CFR Part 70:

> <u>§70.65 Additional content of applications.</u>
> (a) In addition to the contents required by §70.22, each application (for a new Part 70 license) must include a description of the applicant's safety program established under §70.62, including the integrated safety analysis summary and a description of the management measures (including configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements).

From the above, it is obvious that each applicant for a new 10 CFR Part 70 license must:

1. Establish management measures (including configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, **and other quality assurance elements**) to provide continuing assurance of compliance with the performance requirements; and

2. Develop and implement the facility design in accordance with management measures (including configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, **and other quality assurance elements**) to provide adequate assurance that items relied on for safety will be available and reliable to perform their function when needed.

The NRC QA regulatory requirements in the revised 10 CFR Part 70 regulation (quoted above) are very general and nonprescriptive. Therefore, to aid the staff's efforts in any technical review of a license application that might be received in the future under the revised 10 CFR Part 70, the NRC staff has drafted or completed several different Standard Review Plans (SRPs) for different types of fuel facilities. For example, an SRP with QA regulatory guidance for NRC reviewers to use when reviewing the TWRS-P 10 CFR Part 70 license application has been completed and released to the public as NUREG-1702.[66] Chapter 11 of NUREG-1702 addresses Management Measures, and Section 11.3 addresses QA.

In line with the NRC concepts of maintaining safety, reducing unnecessary regulatory burden, improving public confidence, and increasing efficiency and effectiveness of NRC processes, Section 11.3 of NUREG-1702 relies heavily on the industry QA standard, ASME NQA-1-1994. While ASME NQA-1-1994 has separate sections for "requirements" and "guidance," NRC's licensing requirements are existent solely in the agency's regulations. Guidance on how the NRC's regulatory requirements are to be implemented is provided in separate documents such as SRPs, regulatory guides, etc. From an NRC regulatory perspective, none of the parts of

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[66] Nuclear Regulatory Commission (U.S.)(NRC). NUREG-1702, Final Report, "Standard Review Plan for the Review of a License Application for the Tank Waste Remediation System Privatization (TWRS-P) Project." NRC: Washington, D.C. March 2000.

ASME NQA-1 would constitute requirements, i.e., would have the force of law, unless they were incorporated specifically into the applicable regulation (10 CFR Part 70). This is not apt to happen.

Subsection 11.3.5.1, "Acceptance Review," of NUREG-1702 shows that the applicant's/licensee's commitment to implement and maintain its QA program in conformance with the applicable ASME NQA Standard requirements of Parts I and II of ASME NQA-1-1994 or equivalent should satisfy the acceptance review criteria.

Further, Subsection 11.3.5.2, "Safety Evaluation," of NUREG-1702 gives the applicant/licensee the option of:

1. Committing to implement and maintain its QA program in conformance with the applicable ASME NQA Standard requirements of Parts I and II of ASME NQA-1-1994 or equivalent and addressing a relatively short list of regulatory review criteria under the headings of (1) organization, (2) QA function, and (3) provisions for continuing QA

- OR -

2. Not committing to meet the applicable ASME NQA Standard requirements in Parts I and II of ASME NQA-1-1994 but, rather, addressing the same three regulatory review criteria noted above plus a relatively long list (15+ pages) of additional regulatory review criteria as well.

The staff believes that proper implementation of a QA program developed under either Option 1 or Option 2 as allowed in Subsection 11.3.5.2 of NUREG-1702 would meet NRC's QA regulatory requirements in the revised 10 CFR Part 70 and would provide reasonable assurance that the TWRS-P items relied on for safety will perform satisfactorily in service without undue risk to the health and safety of the public or to the environment.

3.10.4 COMPARISON OF QA REGULATORY REQUIREMENTS AND GUIDANCE

The QA regulatory requirements in both NRC's proposed 10 CFR Part 70 and in DOE's QA rule provide for flexibility in the approaches that may be taken to meet the regulations. Although organized and worded differently, the QA regulatory requirements of NRC's revised 10 CFR Part 70 and DOE's QA rule and the corresponding guidance are not inconsistent.

As noted above, NRC's revised 10 CFR Part 70 simply requires that each applicant or licensee establish management measures (including configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements) to provide continuing assurance of compliance with the performance requirements.

Correspondingly, in addition to providing many more detailed QA requirements that correspond to NRC QA guidance, DOE's QA rule requires that (management of) a contractor responsible for a DOE nuclear facility develop, implement, and maintain a QA program acceptable to DOE; that the facility design incorporate applicable requirements and design bases; and that the adequacy of design products be verified or validated.

Although there are many similarities between the DOE and NRC QA requirements and guidance, a QA program could still be developed that meets the QA requirements and guidance of one organization but does not meet the QA requirements and guidance of the other organization. For example, DOE's QA rule requires "assessments" while NRC's guidance refers to "audits." The relationship between assessments and audits is not well defined. Items like this are not expected to be significant issues/problems for TWRS-P because the BNFL Inc. QA program description that the RU approves as meeting DOE QA regulatory requirements, and that BNFL, Inc. implements, is expected to also meet NRC's proposed QA regulatory requirements and guidance.

3.10.5 BNFL INC. QA PROGRAM FOR TWRS-P

BNFL Inc. first submitted a description of its QA program to DOE on November 6, 1996.[67] The NRC provided preliminary comments to DOE regarding that BNFL Inc. submittal on November 20, 1996,[68] but DOE had provided its comments on that submittal prior to receipt/review of the NRC comments.[69]

BNFL Inc. submitted Revision 2 of the description of its QA program to DOE on February 7, 1997,[70] indicating that the revision provided "the agreed upon disposition" of a number of listed items. On February 12, 1997, DOE approved Revision 2 "as the BNFL Quality Assurance Program for Part A activities."[71]

In Revision 3 of BNFL-5193-QAP-01 dated March 27, 1998, BNFL Inc. changed the title of the document from "Quality Assurance Program" to "Quality Assurance Program and Implementation Plan."[72] Revision 3 was a major rewrite of the TWRS-P QA program. It described the BNFL Inc. QA program for activities up to the start of construction and, as the title indicates, the BNFL Inc. plan for implementing its QA program. The NRC provided detailed comments to DOE regarding Revision 3 of the (now-called) Quality Assurance Program Implementation Plan (QAPIP) in a letter dated April 3, 1998.[73]

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67 Bullock, M.J., BNFL Inc., letter to P. Rasmussen, U.S. Department of Energy, with attachment, "BNFL-5193-QAP-01, Revision 0, Quality Assurance Program, November 8, 1996." November 6, 1996.

68 Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to T.R. Sheridan, U.S. Department of Energy, November 20, 1996.

69 Sheridan, T.R., U.S. Depatment of Enegy and C. Bell, U.S. Department of Energy, fax to R. Pierson, U.S. Nuclear Regulatory Commission and J. Spraul, U.S. Nuclear Regulatory Commission, November 20, 1996.

70 Bullock, M.J., BNFL Inc., letter to T. R. Sheridan, U.S. Department of Energy, with attachment "BNFL-5193-QAP-01, Revision 2, Quality Assurance Program, February 7, 1997." February 7, 1997.

71 Sheridan, T.R., U.S. Department of Energy, letter to M.J. Bullock, BNFL Inc., February 12, 1997.

72 Bullock, M.J., BNFL Inc., letter to T. R. Sheridan, U.S. Department of Energy, with attachment "BNFL-5193-QAP-01, Revision 3, Quality Assurance Program, March 27, 1998." March 27, 1998.

73 Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, April 3, 1998.

On April 30, 1998, the NRC staff QA reviewer participated in a DOE-BNFL Inc. meeting at which an April 22, 1998, draft Revision 4 of the BNFL Inc. QAPIP was discussed. The NRC comments regarding Revision 3 of the QAPIP were generally included in the discussion. The meeting resulted in a second draft of Revision 4 of the BNFL Inc. QAPIP dated May 4, 1998. The final version of Revision 4 was submitted to DOE by BNFL Inc. on May 15, 1998.[74] It is dated May 1998.

In Revision 4 of its QAPIP, BNFL Inc. states: "the implementation and maintenance of the QAP (Quality Assurance Program) shall comply with the applicable elements of . . . *Quality Assurance Requirements for Nuclear Facility Applications* (ASME NQA-1 1994a) . . . ." BNFL Inc.'s QAPIP also addresses the regulatory review criteria under the headings of (1) organization, (2) QA function, and (3) provisions for continuing QA specified in Section 11.3 of NUREG-1702. In addition, Revision 4 of the BNFL Inc. QAPIP also addresses the majority of the longer list of additional regulatory review criteria in the Appendix of Section 11.3 of NUREG-1702 as well.

DOE evaluated the May 1998 final version of Revision 4 of the BNFL Inc. QAPIP for TWRS-P using the DOE documents listed in Footnotes 2, 3, and 5. On June 2, 1998, DOE approved the May 1998 final version of Revision 4 of the BNFL Inc. QAPIP for TWRS-P subject to the following two conditions:[75]

1. "The implementing documents and procedures...required prior to start of preliminary design, detailed design, and procurement, shall be issued before the start of those respective phases of project activity."

2. "BNFL shall implement the (QA program)...as approved by the RU,...up to the start of construction."

On May 27, 1999, BNFL Inc. submitted its proposed Revision 5 (as Revision 4a) to its QAPIP,[76] reflecting its annual update of the document and describing its QA program and commitments up to the start of construction. NRC staff comments on the submittal were sent to the RU on June 22, 1999.[77] They were forwarded by the RU to BNFL Inc. on July 1, 1999,[78] as an enclosure to the letter that stated: "These comments (as is customary for such NRC transmittals) are in the public record and are being provided to you for information purposes.

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[74] Bullock, M.J., BNFL Inc., letter to D.C. Gibbs, U.S. Department of Energy and P. Rasmussen, U.S. Department of Energy, with enclosure "BNFL-5193-QAP-01, Revision 4, Quality Assurance Program and Implementation Plan, May 1998." May 15, 1998.

[75] Gibbs, D.C., U.S. Department of Energy, letter to M.J. Bullock, BNFL Inc., with enclosure "DOE Regulatory Unit Evaluation Report of the BNFL Inc. Quality Assurance Program and Implementation Plan, May 1998." June 2, 1998.

[76] Burrows, C., BNFL Inc., letter to D.C. Gibbs, U.S. Department of Energy, with enclosure "BNFL-5193-QAP-01, Revision 4A, Quality Assurance Program and Implementation Plan, May 26, 1999." May 27, 1999.

[77] Pierson, R.C., U.S. Nuclear Regulatory Commission, letter to D.C.Gibbs, U.S. Department of Energy, June 22, 1999.

[78] Gibbs, D.C., U.S. Department of Energy, letter to M.J. Lawrence, BNFL Inc., July 1, 1999.

The Regulatory Unit's comments on the QAPIP, which will include consideration of the NRC comments, will be transmitted separately."

The RU transmitted its comments on the proposed Revision 5 to the BNFL Inc. QAPIP as an enclosure to its letter of July 12, 1999.[79] The letter states: "The RU will use the BNFL response to the comments to complete an Evaluation Report as the basis for approval or disapproval of the QAPIP."

An enclosure to a letter from the RU to the NRC[80] shows the RU disposition of the NRC comments. Of the 29 technical NRC comments, 22 were provided to BNFL as RU comments, one was provided to BNFL Inc. after modification following clarification by BNFL Inc., three were considered to be equivalent to some other RU comment, and the remaining three were transmitted only by DOE's July 1, 1999, letter referred to above. Acceptable justification for not including these last three NRC technical QA comments as RU comments was given in DOE's July 14 letter. All 12 NRC editorial comments were provided to BNFL Inc. as RU comments. Thus, the BNFL Inc. response to the RU comments also addressed the NRC comments except as noted.

Revision 5 of the BNFL, Inc. QAPIP was approved by the RU and issued on April 4, 2000. In June 2000, BNFL, Inc. submitted a proposed revision to address construction as well as design. The RU was planning to review this document in July 2000.

Revision 5 of the BNFL, Inc. QAPIP generally meets NRC QA regulatory requirements for safety of activities under 10 CFR 70. However, three issues regarding adequacy of the interpretation, application or implementation of the QAPIP should be resolved as early in the TWRS-P project as possible. These issues are discussed in detail in Attachment A.19 through A.20 to this report.

1. BNFL Inc. has not adequately identified its commitments or interpretations related to the applicability of specific NQA-1 requirements, either as full and explicit commitments to NQA-1, nor have exceptions to and applicability of the provisions of the document been identified. Implementing procedures for NQA-1 requirements appear not to have been fully and completely developed, and there is no explicit commitment to the records storage and software requirements of NQA-1.

2. The appropriate and adequate implementation by all major team members and subcontractors of QA requirements should be verified prior to or early in the project design activities in order to preclude extensive avoidable retrofits for design, procurement or other project activities.

3. A complete safety categorization of the structures, systems and components (SSCs) has not been fully developed by BNFL Inc. or provided to DOE. Without a list of all SSCs for

[79] Gibbs, D.C., U.S. Department of Energy, letter to M.J. Lawrence, BNFL Inc., July 12, 1999.

[80] Gibbs, D.C., U.S. Department of Energy, letter to R.C. Pierson, U.S. Nuclear Regulatory Commission, July 14, 1999.

each of the safety categories, a logical and systematic graded QA program cannot be developed.

3.10.6 NRC ASSESSMENT OF CURRENT STATUS AND CONCLUSIONS

BNFL Inc. QA commitments that the RU approved as meeting DOE QA regulatory requirements up to the start of construction, generally meet NRC's QA regulatory requirements and guidance. Therefore, the NRC staff concludes that the BNFL Inc. QA commitments for TWRS-P, when properly implemented, can meet the NRC QA regulatory requirements of the revised 10 CFR Part 70 and the NRC QA regulatory guidance in NUREG-1702. Consequently, the staff concludes that the transition of TWRS-P from DOE to NRC regulation and oversight should be relatively smooth in the area of QA.

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4.0 NRC OBSERVATION AND CONCLUSIONS ON THE REGULATORY APPROACH

The basic concept of the U.S. Department of Energy's (DOE's) regulatory approach is that the contractor is responsible for achieving adequate safety, complying with applicable laws and regulations, and conforming with top-level safety standards and principles stipulated by DOE. Consistent with applicable laws and regulations, the contractor is required to tailor the exercise of this responsibility to the specific hazards associated with its activities, and is encouraged to do this in a cost-effective manner that applies best commercial practices. The Tank Waste Remediation System-Privatization (TWRS-P) contractors have the responsibility to identify and recommend to DOE the set of standards, regulations, and requirements necessary to ensure adequate safety. DOE's responsibility is to execute the regulatory process, including authorization of contractor actions and confirmation that the contractor activities are performed safely and within approved limits. The authority of the RU to regulate a TWRS-P contractor is derived from the terms of the TWRS-P contract ("regulate by the contract"). Section 1.4 provides more information on the DOE regulatory approach and supporting documents.

The U.S. Nuclear Regulatory Commission (NRC) staff has participated with DOE in the TWRS-P program for 3 ½ years. NRC staff have observed interactions between DOE and the contractors and the practical effects from the implementation of the regulatory approach. Many of these observations have already been discussed with DOE[81]. An overriding concern has been the way all activities have been and continue to be driven by programmatic concerns, including cost and schedule. Ironically, this emphasis has allowed some areas of the design and safety analyses, such as identification of items relied on for safety (IROFS), to remain relatively nonspecific after the 4 years of design efforts by the Contractor.

4.1 PROGRAMMATIC INFLUENCE UPON DOE REGULATORY ACTIVITIES

The Regulatory Unit (RU) is part of the DOE Richland Operations Office (RL) and reports to the manager of RL (i.e, at the time of contract termination in June 2000). The RL organization reports to DOE/Environmental Restoration and Waste Management (EM) in Washington, D.C. Initially, the DOE manager for TWRS-P also reported to RL. In 1998 the DOE Office of River Protection (ORP) was created to handle the management of TWRS and related activities, including TWRS-P. ORP directly reports to DOE-EM with matrix activities to RL. ORP manages and administers the TWRS-P contracts. Thus, the RU and the ORP are both parts of the DOE-EM organization and influence is unavoidable. As noted by a DOE external review of the RU (see Footnote 12, page 28), true regulatory independence may not be achieved by the current approach. Finally, RL is a party to the Tri-Party Agreement (TPA), which emphasizes schedules with milestones for remediation of the Hanford site (see Chapter 1.0). This has the unintended consequence of emphasizing schedule for the RU—"In the context of contract-

81 Leach, M.N., U.S. Nuclear Regulatory Commission, letter to D.C. Gibbs, U.S. Department of Energy, "Potential Critical Technical Issues for Construction Authorization Resolution," January 21, 2000.

based regulation, the regulator has an incentive to conduct the necessary regulatory activities in such a manner that the project schedule is not unduly delayed.[82]"

The DOE approach regulates by incorporating top level standards and requirements into the TWRS-P contracts. Thus, the RU regulates "by the contract." The TWRS-P contract (Main Reference 21) contained many milestones, interactions, and deliverables for program purposes and reviews, but only three for regulatory purposes: the Design Safety Features (DSF), the Topical Meetings, and the Construction Authorization Request (CAR). In addition, the RU made numerous statements that they could only regulate to the contract and frequently referred to the schedule of deliverables it contained. The actual program contract officers for DOE are in the ORP, and the Contractors appear to focus more on feedback from the program side than from the RU/regulatory side.

DOE has encouraged the Contractor to pursue an approach with a large processing facility containing several melters (the largest proposed melters for radwaste use in the world), close to two million gallons of liquids, and potentially tens of megacuries of activity. This emphasis on a much larger facility accrues from program issues and TPA commitments. Regulatory and safety issues associated with a much larger facility do not appear to have been considered. In contrast, the approaches at Defense Waste Processing Facility (DWPF) and the West Valley Demonstration Project (WVDP) used several years of melter testing and extensive experimental studies and pilot plant testing. The concept of a smaller pilot facility was abandoned during Phase IA at Hanford. Given the many unknowns associated with the different wastes at Hanford, it would seem an emphasis on the large facility could encounter numerous problems during the processing of actual waste that could be better addressed in a pilot scale facility.

Further evidence of programmatic influence upon the regulatory activities are found in the review schedules. The reviews of significant deliverables would sometimes overlap or conflict. Frequently, the DOE and/or RU would only allocate a very short, limited time for conducting significant reviews of major deliverables. A typical review time would be 2 or 3 weeks for multivolume submittals and is insufficient for adequate review. For example, the RU allowed two weeks for the review of the Initial Safety Analysis Report (ISAR) in 1998 and, even though it was given adequate notice, did not appear to actively consider additional NRC comments submitted after the 2-week period. The Firm Fixed Price (FFP) submittal totaled seven boxes (of interest for design and safety—some 25,000 pages); the active review lasted about 3 weeks. The review letter from the RU was signed out inside of 1 month and consisted of only four pages[83].

Sometimes cost and schedule issues also influenced regulatory operations, including reviews. Terms such as "cost effective," "return on investment," and "impact to schedule" would frequently arise in reports and at meetings discussing safety and regulatory matters. For example, there was a discussion at the June, 2000 Topical Meeting about the relative costs of

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[82] Gibbs, D. Clark. <d_c_clark_gibbs@rl.gov> to W.J. Pasciak, U.S. Nuclear Regulatory Commission, "Former Regulatory Transition Plan," with attachment "RL/REG-99-01, Issues Related to the Potential Regulatory Transition of TWRS-P" November 1998. (December 15, 1998).

[83] Gibbs, D.C., U.S. Department of Energy, memorandum to R.T. French, U.S. Department of Energy, "REG:RAG/00-RU-0395, Review of BNFL Inc. (BNFL) Part B-1 Facility and Process Design Deliverables by the Office of Safety Regulation (RU)," May 25, 2000.

ventilation controls and instrumentation. On many occasions, there was an implication that regulatory reviews were not allowed to impact cost and schedule.

Finally, as already noted, the DOE has terminated the TWRS-P contract and is transitioning to a Management and Operations (M&O) contract for TWRS. Few of the recent decisions and activities on the contracts have actually mentioned safety (see the ORP website via www.hanford.gov). The Request for Procurement mentions regulation by DOE but offers incentives of 20-30 percent for cost savings; no incentives for safety are mentioned.

4.2 INADEQUATE MAINTENANCE OF DESIGN AND AUTHORIZATION BASIS DOCUMENTS

Throughout most of the design effort (about 2 years), the design and safety teams of the Contractor have worked quasi-independently. The design and authorization basis documents have not been updated (prior to contract termination, the plan was to issue updates in late 2000) and amendment requests have only recently started to be received from the Contractor. Changes in fundamental aspects of the design have occurred in this time period without regulatory review. This was formally noted after contract termination[84] although the changes were made about 18 months earlier. DOE subsequently approved these changes after a qualitative evaluation[85].

The contractor has selected and proposed standards and regulations for TWRS-P, which have subsequently been approved by the RU. These, combined with the design, became the "authorization basis." A formal change process has been identified for the standards and regulations. However, in practice, it was not clear if these standards and changes are being adequately maintained and updated by both the regulator and the Contractor, and if they are being adequately complied with by the design teams and reviewers. For example, during Part B-1, an inspection of the authorization basis was twice postponed because it was known the Contractor was not following procedures and maintaining the design/authorization basis, and would not pass the inspection. While a corrective action conference was ultimately conducted (see Section 3.1), this occurred much later in March 2000—some 19 months after the start of Part B-1. As a further example, in response to an NRC inquiry about revision dates for the Safety Requirements Document (SRD), the DOE advised that its version of the SRD was not a controlled copy and is updated by revised pages rather than a complete document (see Footnote 20 on page 41). In a recent design review on "Pretreatment Ion Exchange Systems" in June 2000, the discussion indicated that the design and the design team had not considered the standards and requirements in the authorization basis—this was overlooked by the design

[84] Gibbs, D.C., U.S. Department of Energy, letter to P.O. Strawbridge, BNFL Inc., "Contract Number DE-AC27-96-RL13308 - Regulatory Unit (RU) Partial Approval of Authorization Basis Amendment Request (ABAR), ABAR-W375-00-00014, Rev. 0, Part A Hazard Analysis Report (HAR) Significant and Bounding Hazard Evaluation and Initial Safety Analysis Report (ISAR) Fundamental Aspects of Design," 00-RU-0455. July 6, 2000.

[85] Gibbs, D.C., U.S. Department of Energy, letter to P.O. Strawbridge, BNFL Inc., "Contract Number DE-AC27-96-RL13308 - Regulatory Unit (RU) Approval of Authorization Basis Amendment Request, ABAR-W375-00-00014, Rev. 0, Part A HAR Significant and Bounding Hazard Evaluation and ISAR Fundamental Aspects of Design," 00-RU-0529. August 10, 2000.

review report[86]. In contrast, the NRC uses a formal process with public participation to codify regulations and establish guidance and licenses. For changes and deviations in the standards used or proposed by licensees, the NRC would conduct a comprehensive review of the application and documentation incorporating the change.

The DOE and the RU frequently accept the standards and regulations proposed by the contractor with a limited independent review of its applicability to the proposed design. It is not clear if DOE is independently assessing the design's compliance with the selected standards and authorization basis. The burden is placed upon the TWRS regulator to analyze and approve or disapprove the standard. The NRC would conduct a thorough review of any applicant/licensee from NRC requirements and/or NRC-endorsed industry codes and standards. More of the burden is placed upon the license applicant. Significant changes might require NRC reviews considerably longer (up to a year or more) than some of those observed during this program and would be based on a more complete level of design.

The design and authorization basis—the license basis—are spread over several different documents of differing vintages and designs. Changes, including modifications in approach, are not always well documented or substantiated. For example, the FFP contract is more detailed than the ISAR, yet it lacks some features (e.g., some tanks and pumps) and adds others (e.g., an extra ion exchange column), without explanation. Various capacities are mentioned in other documents (i.e., 1x, 2x, and 4x flow rates) without a clear explanation of the bases for some portions of the proposed facility and equipment handling much greater quantities of materials and activity. The authorization basis has not been well defined in quantitative parameters, including the flow rate level. This has important ramifications for the safety reviews—obviously a scenario with a release at a 4x flow rate is likely to have a greater consequence than one at a 1x flow rate, and this may influence severity levels and safety controls.

4.3 USE OF A RISK-BASED APPROACH

The use of a risk-based approach to the design and risk-based analyses were used as the basis for the Integrated Safety Management (ISM) process, which includes hazards identification, consequence estimation, and control mitigation with limited additional considerations. This is essentially a completely fluid process without a basal level of requirements and, as presently practiced, does not consider unknowns, uncertainties, errors, proven practices, future plans, and experience. As practiced in TWRS-P, there appears to be more emphasis on the process and less on the results.

The ISM approach at TWRS-P is a circular process (see Section 1.4). A clear, central concept of ISM is that the contractor should tailor the design and safety requirements to the specific hazards of the activities and operations at a facility. DOE/RU policy endorses tailoring via the following process:

1. Identify applicable requirements.
2. Define the scope of the work or operations to be analyzed.

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86 Gilbert, R.A., U.S. Department of Energy, memorandum to Regulatory Unit Staff, U.S. Department of Energy, "Design Review Report—April-June 2000 Design Reviews," August 4, 2000.

3. Analyze the hazards.
4. Propose, analyze, select, and implement controls.
5. Perform the work or operations (does not apply at the design stage).
6. Assess, feedback, and improve/modify (as appropriate).

As the ISM reviews cycle, the concept is for the safety controls to become tailored to the hazards. However, an unintended consequence of this approach is the potential elimination of margins, conservatism, and defense-in-depth in the design and safety bases because minimal requirements are not incorporated. This can obscure minimum standards and performance requirements commonly found to be effective from experience and lessons-learned. In addition, as the ISM "cycles," assumptions about source terms, releases, etc., are challenged and may be reduced. In practice, the assumption can sufficiently reduce consequences to place a scenario(s) into less severe categories. Since the ISM approach focuses on the higher risk scenarios, once a scenario's consequence is reduced, the assumptions may not be revisited. Without a minimal level of requirements, this circular logic may result in fewer safety controls and more risk from the proposed facility. The preliminary nature of the design and Part B-1 further compounds these concerns as more conservatism would seem to be needed when less design information is available. DOE has also experienced difficulty communicating the ISM approach to the contractors. For comparison, the NRC uses a risk-informed, performance-based approach with defense-in-depth, appropriate levels of conservatism, and a minimum set of standards and requirements that are codified in the regulations.

The TWRS-P regulatory approach has requirements for reviewing conservatism and defense-in- depth (DID - see CAR guidance in Chapter 7.0, Main Reference 14). DOE most recently has expressed concerns about too much conservatism, in public statements on the FFP submittal[87]. In regards to the same FFP submittal, DOE has also noticed the level of conservatism and design margins are not mentioned and had concerns about spares and redundant equipment and instrumentation. These concerns about conservatism and redundancies are further elaborated upon in a more detailed letter[88]. As the ISM has been practiced, most high severity scenarios have just two safety control approaches.

The DOE and RU acknowledge that there are unknowns and uncertainties associated with the program and these should be considered in the reviews of submittals. However, as a practical matter, little effort has been given into defining this further and only limited feedback has been provided to or received from the contractors. At the present time, the focus continues to be upon best-basis and average conditions for many of the analyses. The use of average conditions without addressing uncertainties and incorporating reasonable conservatism and DID is not likely to be acceptable to the NRC. Furthermore, there does not appear to be a realization of future effects. These include site changes (e.g., other privatization initiatives, relaxation of security) and facility effects (e.g., higher flow rates desired, wear and aging of structures, systems, components (SSCs), erosion/corrosion effects, process changes).

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[87] *Inside Energy*, May 15, 2000.

[88] Gibbs, D.C., U.S. Department of Energy, letter to P.O. Strawbridge, BNFL, Inc., "Regulatory Unit (RU) Review of the BNFL Inc. Part B-1 Facility and Process Design Deliverables, (FFP Submittal)," 00-RU-418. June 2, 2000.

The RU expects the facility design to be consistent with a risk level of circa 1E-5/yr to the worker and circa 1E-6/yr to the public. At face value, these are approximately comparable to the NRC policy statements on risk and lower than the risk levels implied by the new 10 CFR Part 70. However, the public doses are estimated at a minimum distance of the site boundary (12 km to the receptor) and, sometimes, further. Nonconservative values are used for input data. The application of ALARA (as low as reasonably achievable) to the design appears limited. The RU has allowed the use of a chemical safety standard that increases the potential risk and allows the existence of a relatively high probability, high consequence event. The NRC would likely ask for more substantiation of the values and approaches used with appropriate levels of conservatism and redundancy.

The lack of a clear approach contributes to confusion regarding the level of conservatism, particularly for a preliminary design. Using the NRC SRP for TWRS (NUREG-1702) as a guide, it is unlikely that the current design would have adequate conservatism and DID if it were undergoing a licensing process with the NRC.

4.4 LIMITED USE OF NRC REGULATIONS AND GUIDANCE

The DOE regulation of TWRS-P primarily uses documents and guidance written by DOE and its contractors specific to Hanford and the tank wastes (see Section 1.3). Limited use is made of NRC guidance and regulations apart from the principles of good regulation. For example, the NRC has revised its 10 CFR Part 70 regulations which would be used if the TWRS-P were to transition to NRC regulation. The NRC has published in final form a standard review plan for TWRS-P facilities (Main Reference 15). These can be used for either one-step (combined construction and operating) or two-step (construction first, followed by a separate operating submittal) licensing. However, the TWRS/River Protection Project does not plan to use either in the regulation of these facilities. In addition, there is a desire to avoid nuclear reactor-related regulations, guidance, codes, and standards, without fully evaluating the applicability and appropriateness of these reactor areas of review. Given the highly radioactive nature of the materials that would be in the proposed TWRS-P facility, it would seem that some reactor regulations and guidance might be applicable.

There are also differences in the NRC approach. While the NRC does not usually participate in discussions and meetings with potential licensees at a preliminary design level, it occasionally does (e.g., the proposed mixed oxide fuel facility). The NRC does not develop standards via a contractual, licensee developed process. The NRC process for developing regulations invites stakeholder participation (e.g., the recent 10 CFR Part 70 revisions) and, once the regulations are developed, approved and promulgated, the NRC expects all of the affected licensees to abide by the regulatory requirements. The related guidance might include values or methods for determining source terms and release fractions, an area where there have been significant variations in TWRS-P—for example, the RU initially suggested the use of lower radionuclide concentrations after encouraging the contractor to go with higher ones. The NRC generally does not develop customized regulations and standards for each licensee. This approach helps to maintain consistency in regulatory and safety matters.

Unlike many other DOE programs, there did not appear to be any routine reports (monthly, quarterly, or annual) that definitively discussed the current design, safety features, and the evolution thereof. The standard approach of issuing a draft followed by a final version (after comment resolution) a few months later has not been followed for major reports. For example, the FFP submittal represented the first complete design "report" after the ISAR, corresponding to a chronological spacing of some 28 months. Revisions and updates of major deliverables are not timely; the first SRD revision occurred one year after the initial version, and a revised Hazards Analysis Report will occur 3 years after the original version. Similar communication issues exist with Safety Analysis Reports—the time period between the ISAR and the CAR/Preliminary Safety Analysis Report will be at least 3 years. The RU has closed items from previous reviews after discussions with the contractor. However, it is not clear if these are actually closed in the usual sense of the word (i.e., fully addressed now) or adequately addressed in an SER-like (safety evaluation report) method. Frequently, "closure" is used to denote agreement on an approach to address the issue, not necessarily closure itself. Sometimes a clear course of action or a commitment by the contractor is not apparent. The DOE program managers in ORP did not appear to actively communicate with the RU on a routine basis and at a functional level. The NRC has found that clear documentation, communication, and tracking are beneficial for regulatory reviews.

DOE has used subcontractors extensively for reviews. For example, approximately 50 percent of the RU budget is due to subcontractors. Relatively few people in the DOE/RU/contractor environment appeared to have a background compatible with greenfield design and regulation of a new process and facility. The subcontractors have performed significant roles in the reviews for large submittals and have developed and written guidance. At times, DOE has performed more in an oversight role over their subcontractors than in some of the reviews. In contrast, the NRC tends to rely less on subcontractors.

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5.0 POTENTIAL ISSUES FOR TRANSITION TO U.S. NUCLEAR REGULATORY COMMISSION REGULATION

The U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) have previously discussed issues related to the potential regulatory transition of Tank Waste Remediation System-Privatization (TWRS-P) to NRC regulation in the near future. Many of these issues are summarized in Main Reference 17 and have been discussed between the DOE and NRC over the 3 year length of the program. DOE is converting the contracts to an Management and Operations (M&O) arrangement for TWRS, which has been renamed the River Protection Project-Waste Treatment Plant (RPP-WTP). The NRC staff believes that enabling legislation from Congress is required for the NRC to regulate either a privatized TWRS-P facility or an RPP-WTP, and that any resulting issues are resolvable. From the viewpoint of the NRC staff, most of the issues would be addressed by the legislation that enables NRC regulatory authority over the TWRS/WTP facilities or NRC external regulation of DOE facilities, and by continued refinement and detailing of the proposed facility designs. The remaining issues relate to DOE programmatic activities and not regulation. The following sections summarize the issues and the preliminary assessment of the NRC staff; Attachment D provides a discussion of the issues in more detail.

5.1 EMERGENCY PLANNING

The TWRS-P Contractor defined an approach in accordance with DOE guidance. There were several issues related to the integration of the TWRS-P emergency plan (EP) with the Hanford site EPs and organizations (e.g., the Hanford Fire Department) which were being worked upon at the time of contract termination. It is anticipated that the transition and subsequent new M&O contractors will address these integration issues. From the point of view of the NRC staff, if NRC were to regulate the RPP-WTP facility, it would be expected that the integration issues will be fully resolved, communication and responsibilities will be clearly defined, and the WTP facility EP manager will have the authority to direct the Hanford site EP as necessary based upon the integrated safety analysis (ISA).

5.2 REGULATORY AUTHORITY AND CLASSIFICATION OF RADIOACTIVE WASTES

As designed by the Contractor, the TWRS-P facility would have generated vitrified low activity waste (LAW) and facility wastes, such as failed melters and high efficiency particulate air (HEPA) filters. Long term storage of these wastes was planned at the Hanford site. If the decision were made for disposal at the Hanford site, existing waste management and disposal areas would be used, subject to the wastes meeting the applicable disposal unit's waste acceptance criteria. These disposal facilities are regulated by DOE for radioactive components and by the State of Washington for hazardous constituents. If the NRC were to regulate RPP-WTP, the vitrified LAW and facility wastes would be regulated by the NRC for their radioactive components while they are located at the facility. The NRC would likely designate the vitrified LAW and facility wastes as incidental wastes, i.e., incidental to high level waste (HLW) processing. After the contractor has returned the wastes to DOE, regulation of the radioactive component would revert to DOE provided incidental waste criteria are met. The NRC and DOE have discussed the handling of incidental wastes and criteria several times on other HLW programs; basically, the incidental waste was considered to be suitable for near-surface disposal if 10 CFR Part 61 criteria for low level waste (LLW) are met. RPP-WTP incidental wastes are likely to meet those criteria, and hence, the issue should be easily resolved.

5.3 SAFEGUARDS AND SECURITY

The physical protection requirements for various classifications of nuclear materials are very similar between the DOE and the NRC. Both recognize the differences in the quantities and attractiveness characteristics of the materials in the application of the requirements. In addition, for special nuclear material (SNM) at a TWRS facility, the facility design (multiple confinement layers and cells), the dilute concentrations of SNM, and the highly radioactive nature of the matrix (waste) containing the SNM contribute to its physical protection. DOE has categorized the tank materials as waste and removed it from the safeguards program. The NRC believes that operations at the proposed TWRS facilities and wastes from specific tanks may result in the accumulation or *de facto* separation of SNM, and, thus, might require safeguards. Consequently, the NRC has the position that the issue should be revisited once the design is further along (say at the Construction Authorization Request/Preliminary Safety Analysis Report (CAR/PSAR) stage) and when more operating details are available (for example, at the Operations Authorization Request stage). Given the similarity between NRC and DOE safeguards requirements, the NRC does not see this as a significant issue.

5.4 REGULATION OF EMISSIONS

Under the current regulatory approach, air emissions from the facility are regulated by the State of Washington. If the NRC were to assume regulatory authority for TWRS/WTP, the NRC would regulate portions of air emissions. Since there is precedence for this approach (both the State and NRC regulate emissions at the Siemens facility), the NRC staff does not view it as a significant issue.

5.5 CO-LOCATED WORKER

Section 3.4.2 presents more information on the co-located worker (CLW). In summary, the DOE is using the concept of CLW for the proposed TWRS/WTP facilities; essentially the CLW is allowed the same dose and risk limits as the workers and all members of the public, visitors, etc., to the site are considered "general employees" and are allowed to accumulate occupational exposure. For TWRS-P, DOE plans to use a large portion of the Hanford site for this purpose, which typically results in minimum distances to members of the public (i.e., with lower dose and risk limits) of approximately 10 miles. In contrast, the NRC focuses on the concept of "controlled area" by the facility operator; at NRC facilities, this is usually associated with a fence although the controlled area can extend beyond the fence and even beyond the site boundary. Thus, the NRC approach usually corresponds to a shorter distance (usually 100-200 meters) to the public for accident analysis purposes, which can translate into more items relied on for safety (IROFS). However, the key concept is the TWRS/WTP operator's authority to exercise control over the Hanford site emergency plans; if such authority is granted by DOE to the TWRS/WTP operator, then the "controlled area" concept may be satisfied, and the issue is moot.

In addition, the dose limits require comparison. Under the TWRS-P/privatization regulatory program, the dose limits were 25 rem for the worker, CLW, and the public for highly unlikely events (the public had a target dose goal of 5 rem). In DOE, highly unlikely corresponds to the frequency range of 1E-4/yr to 1E-6/yr. From the perspective of the NRC staff, the revised 10 CFR Part 70 and standard review plan (SRP) have a worker dose limit of 100 rem and a public dose limit of 25 rem for high consequence events, and corresponding limits of 25 rem and 5 rem for intermediate consequence events. High consequence events are to be rendered highly unlikely (1E-5/yr or less in frequency) by safety controls, and intermediate consequence events are to be rendered unlikely (in the 1E-2/yr to 1E-5/yr frequency range) by safety controls. Thus, at face value, the DOE limits are more restrictive although the fuel cycle SRP[89] does allow grading of the frequency limit in inverse proportion to the magnitude of the consequences and this could result in the limits overlapping. Again, the NRC staff considers this to be a resolvable issue.

As an aside, the NRC notes the DOE regulatory approach should consider future site changes planned that are likely to reduce the distances to the public for accident evaluation purposes (see Issue A.13 in Attachment A).

5.6 WASTE OWNERSHIP

As part of the privatization contracts, DOE retained ownership of the waste materials. Part 70 of 10 CFR does not require transfer of ownership. Thus, assuming the new contract(s) still require ownership of the wastes by DOE, the NRC staff do not believe there is an issue.

5.7 PAYMENT FOR TRANSITION TO NRC REGULATION AND NRC REGULATORY OVERSIGHT

NRC involvement in TWRS-P has been funded via a line item in the budget. Future regulation of a TWRS/WTP facility by the NRC would require legislation that would also identify the funding mechanism, be it by line items, fee collection, DOE payments, or a combination thereof. It is anticipated that the NRC regulatory costs would be a small fraction of the actual DOE expenditures to the contractors on the tank waste programs. The NRC staff does not see this as a significant issue.

5.8 TRI-PARTY AGREEMENT

DOE is a party to the Tri-Party Agreement (TPA) and is responsible for TPA commitments. The NRC staff would expect the NRC to remain a non-party to the TPA consistent with other regulators (e.g., Defense Nuclear Facilities Board (DNFSB), Washington Department of Health (WDOH)). The NRC staff does not see this as a significant issue.

5.9 DOE STOP WORK AUTHORITY

The DOE currently has stop work authority for safety concerns under the contract. The NRC staff anticipates both the NRC and DOE would have stop work authority for safety concerns if regulatory transition occurred, and the staff does not consider this to be a significant issue.

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[89] Nuclear Regulatory Commission (U.S.)(NRC). NUREG-1520, "Draft Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." NRC: Washington, D.C. 2000.

5.10 DOE SAFETY OVERSIGHT AFTER REGULATORY TRANSITION

The DOE currently is the regulator for the proposed TWRS/WTP activities. After transition, the NRC would be the regulator. The NRC staff anticipates the DOE would want to perform some safety oversight activities in a manner analogous to a corporate headquarters unit providing oversight to an operating facility. The NRC staff does not consider this to be a significant issue,

5.11 APPLICATION OF NRC (10 CFR PART 2) HEARING REQUESTS

The potential regulation of the proposed TWRS/WTP facilities by the NRC would probably invoke the NRC Rules of Practice (i.e., 10 CFR Part 2), which allow for public hearings. Hearings require scheduling and can take time. However, licensing under 10 CFR Part 70 would focus hearings on more specific issues and be relatively expeditious. In addition, legislation authorizing NRC regulation of TWRS/WTP might dictate a schedule for hearings. Furthermore, if the facility is well along into construction or even starting operations at the time of transition to NRC regulation, a certification process (for existing facilities) may be more appropriate. Certification would allow the activities at the plant to continue while the hearings and other reviews are completed. Consequently, whether the route is licensing, Congressional mandate, or certification, the NRC staff does not foresee a significant schedule impact from the hearing process, and, thus, does not consider this to be a significant issue.

5.12 CONTRACTUAL OBLIGATIONS FOR FEED DELIVERY

DOE will continue to deliver radioactive feed material to the Contractor. Presumably, DOE will agree to abide by the direction of the NRC-regulated Contractor as to the rate and characteristics of this feed material. The NRC staff expects there will be specifications and safety requirements for the wastes and the plant operations. If the wastes cannot be blended, then an amendment process could be pursued by the licensee (an amendment process has already been implemented at the gaseous diffusion plants (GDPs). This is not an issue for regulatory transition.

5.13 COMMUNICATION PLAN WITH STAKEHOLDERS

The NRC anticipates that the existing DOE communication plans (e.g., with the Hanford Advisory Board) would continue during and after transition to NRC regulation, as they are separate from the regulatory processes. In addition, the NRC also maintains public communication and openness as part of its regulatory practices. Therefore, the NRC staff does not consider this a significant issue.

5.14 COST BENEFIT OF NRC REGULATION

The NRC staff considers NRC regulation of TWRS/WTP facilities to be a national policy issue and not decided by cost/benefit analyses. The NRC staff does not consider this a transition issue.

5.15 ROLE OF THE DEFENSE NUCLEAR FACILITIES SAFETY BOARD

The NRC has determined that legislative authority is required for it to regulate TWRS/WTP facilities. As part of such legislation, the NRC staff anticipates that regulatory roles will be clearly defined and based upon precedent at the enrichment and spent nuclear fuel (SNF) storage facilities that the NRC regulates at DOE sites; the staff does not expect a role for dual NRC/DNFSB oversight of TWRS/WTP. The NRC staff does not consider this an issue.

5.16 PRICE-ANDERSON INDEMNIFICATION

The NRC anticipates the legislation enabling NRC regulation will define any Price-Anderson indemnification requirements. The NRC staff does not consider this an issue.

5.17 TIMING TO AVOID DELAYS FROM REGULATORY TRANSITION

DOE wishes to select the timing for transition so as to reduce the potential for disruption to a minimum. Some timing considerations tend to conflict. In programmatic terms, it would be beneficial for DOE that transition occurs sometime after full and confirmed operation so that it could be assured that the plant meets its specifications. On the other hand, to minimize potential retrofit changes in meeting NRC's regulatory expectations, it might be preferable to make the transition as soon as possible. (This can also be accomplished by resolving all regulatory differences prior to construction and transitioning later.)

The NRC staff believes that there are three alternatives for the timing of regulatory transition: 1) immediately, 2) overlapping, or parallel, regulatory transition process during design, or 3) regulatory transition during operations. Alternatives one and two correspond to licensing of the vitrification plant, for which rules and guidance already exist. Alternative 3 represents a certification route.

The NRC staff believes the legislation establishing NRC regulation of TWRS/WTP will define the timing for transition.

5.18 DOE FINANCIAL INTERESTS

DOE has expressed concerns about protecting its financial interests in the vitrification facility. DOE has already invested several hundred million dollars in the Phase I activities. The completion of the design and construction activities in Phase I may amount to another $4 Billion. Thus, DOE has considerable financial interests in the program. The NRC has no financial interests to protect. The NRC staff does not consider this a regulatory transition issue.

5.19 QUALITY ASSURANCE

The DOE TWRS Quality Assurance (QA) program requirements and the TWRS-P contractor's DOE-approved QA programs may need to be modified in order to facilitate the transition to NRC regulation. The NRC staff believes the differences in QA requirements are small and the adequacy of the QA program really depends upon its implementation. The NRC staff does not consider this a significant transition issue.

5.20 IMPACTS ON OTHER HANFORD SITE FACILITIES

DOE is concerned that there may be impacts upon non-TWRS/WTP facilities at Hanford after regulatory transition to the NRC. The technical impacts to non-TWRS-P facilities of TWRS-P regulatory transition would have to be identified, evaluated, and addressed on an individual basis as such issues arise. Coordinating the transition to external regulation of the tank farm storage and vitrification facilities will likely reduce the impacts to the non-vitrification portion of the tank farms, which is the non-TWRS/WTP facility most susceptible to these impacts. The NRC staff does not consider this an issue for regulatory transition.

5.21 RESPONSIBILITY FOR OCCUPATIONAL SAFETY

For its facilities, DOE has regulatory authority for occupational safety. For NRC licensed facilities, the Occupational Safety and Health Administration (OSHA) has such authority. If the TWRS/WTP transitioned to NRC regulatory oversight, OSHA may assume regulatory authority for occupational safety. The NRC staff anticipates that the enabling legislation for NRC regulation would clearly define the responsibilities.

5.22 TAILORING OF REQUIREMENTS

The NRC and DOE use different approaches for the tailoring of requirements for safe mission performance. This issue concerns the different approaches used by DOE and NRC for establishing requirements for achieving adequate safety and reconciliation of any differences in the approaches to facilitate seamless transition from DOE to NRC regulation, should transition occur at a future date. As discussed in more detail in Attachment D, even though the DOE approach to establishing requirements is somewhat different that the NRC approach, the requirements resulting from either approach will likely achieve adequate safety, and significant reconciliation may not be necessary. The NRC staff believes this issue is not significant and disappears if regulatory transition occurs in the near-term.

5.23 ACHIEVEMENT OF ADEQUATE SAFETY

DOE and NRC conclusions on achievement of adequate safety may be different and need reconciliation to facilitate possible future transition of the TWRS/WTP project to NRC regulation. The DOE regulatory approach for TWRS-P safety (radiological, nuclear, and process) places on the TWRS-P Contractor the responsibility to achieve (a) adequate safety for the workers and the public (b) comply with applicable laws and legal requirements, and (c) conform to DOE stipulated top-level safety standards and principles.[90] Within this regulatory framework, the TWRS-P Contractor identifies the work needed, evaluates the associated hazards, selects appropriate standards, and justifies the adequacy of the selected standards.[91] The NRC has expressed the following concerns about the process for TWRS-P:

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[90] Department of Energy (U.S.)(DOE). RL/REG-97-10, Rev. 1, "Radiological, Nuclear, and Process Safety Regulation of TWRS Privatization Contractors Regulatory Plan." January 1998.

[91] Department of Energy (U.S.)(DOE). DOE/RL-96-0004, Rev. 0, "Process for Establishing a Set of Radiological, Nuclear, and Process Safety Standards and Requirements for TWRS Privatization." February 1996.

1. Safety requirements were proposed and conditionally approved during the pre-conceptual design phase of the facility. This attempt to reach judgment on adequate safety was performed prior to the majority of design information being available.

2. For both regulatory processes, the determination of adequate safety is somewhat subject to engineering judgment, and as such, should transition occur, additional requirements and potential backfits could be needed.

The NRC staff believe this is a technical issue that requires additional design effort by the contractors.

5.24 EFFECT OF SCHEDULE DELAYS ON THE OVERALL RISK

DOE has expressed interest in balancing the risks posed by continued operation of the tanks with the risks posed by operation of the vitrification plant, regardless of the regulatory framework, as a way of avoiding schedule delays. In theory, the requirements could be different (and less) against a baseline of an existing hazard which could be reduced by the operation of the facility. From the viewpoint of DOE, early startup, high availability, and high capacity operation of the vitrification plant could serve to reduce the risks posed by the tanks, even if regulatory requirements on the vitrification plant were relaxed.

NRC requirements have been established to minimize risks of operation of a facility relative to a baseline of no-operation, even though NRC can consider balancing risks within a facility. However, NRC operates to codified regulations (10 CFR Part 70) and, while provision for risk benefits and balancing can be made via the risk-informed, performance-based approach, it can be difficult to manage and balance the different risks on a quantitative basis with limited information on the design and safety features of the TWRS/WTP facility. Regulatory requirements could be administered with recognition of the real risks from the degrading tanks as baseline risks.

The NRC staff anticipates that the scope of NRC regulation—the TWRS/WTP facilities, tank farms, other Hanford facilities—will be defined in the enabling legislation. A single regulatory entity offers consistency and the potential to consider total risks from tank waste management. However, absent specific design and quantitative analysis, it is not possible to assess balancing total risk for the tank waste/HLW related systems at Hanford and achieving adequate assurances of safety is likely to be based upon individual facility analyses.

5.25 RESOURCE ALLOCATION ACROSS TWRS-RELATED FACILITIES

DOE has expressed concerns about balancing the resources available for the safety features of the TWRS/WTP (the vitrification facilities in particular) across the entire TWRS complex. The NRC is concerned with adequate assurances of safety for the workers, the public, and the environment. Once safety and regulatory requirements are met for TWRS/WTP, resource allocation is an internal DOE matter. The NRC staff expects the scope of NRC regulation to be defined in the enabling legislation.

5.26 BACKFIT/RETROFIT

Changes may be required as the TWRS/WTP facility transitions to NRC regulation. The TWRS/WTP facility could be subjected to new or modified requirements imposed by NRC if regulatory transition occurs. These new or modified requirements may require that the TWRS/WTP facility be modified. Terms such as "backfit" and "retrofit" are commonly used to describe modifications that are made in response to new or modified requirements. The NRC anticipates that some backfits/retrofits may be necessary upon transition of the TWRS/WTP to NRC regulatory authority. However, the NRC staff expects these to be minor because the facilities would represent new design and construction with complete documentation and the NRC has participated with DOE and their contractors in Phase IA and Phase IB-1 of the program. Regulatory transition of other TWRS activities might involve older facilities with less documentation and could result in more significant requirements.

5.27 SCOPE OF NRC REGULATION

DOE has discussed the scope of NRC regulation, including facilities beyond the immediate TWRS/WTP facilities.

Cases exist or will exist on DOE sites where NRC regulates one facility and DOE regulates other facilities, although these have much simpler interfaces than at TWRS. The TMI-2 Independent Spent Fuel Storage Installation (at the DOE Idaho Site) is an example of a case where NRC will license and regulate a new facility that will receive waste from an existing facility, specifically the Test Area North (TAN) facility. In this case, DOE regulates current TAN facility spent fuel pool storage operations and will regulate dry storage cask loading operations performed at the TAN facility. NRC regulation will begin when the fuel leaves the TAN facility boundary in an approved transportation cask.[92] In another case, at the Portsmouth and Paducah GDPs, certain operations have been certified by NRC while others remain under DOE control. This shared site arrangement required that regulatory boundaries be established to ensure a clear understanding of responsibilities for regulatory oversight and the rules governing particular portions of the plants.[93]

The NRC staff anticipates that the scope of NRC regulation—the TWRS/WTP facilities, tank farms, other Hanford facilities—will be defined in the enabling legislation. A single regulatory entity offers consistency and the potential to consider total risks from tank waste management. However, absent specific design and quantitative analysis, it is not possible to assess balancing total risk for the tank waste/HLW related systems at Hanford and achieving adequate assurances of safety is likely to be based upon individual facility analyses.

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[92] Department of Energy (U.S.)(DOE). Section 1.2, Rev. 0, "License Application for the Idaho National Engineering Laboratory Three Mile Island Unit Two Independent Spent Fuel Storage Installation, DOE Idaho Operation Office." DOE: Washington, D.C. October 1996.

[93] Buhl, A.R., T. Murley, G. Edgar and D. Silverman. "NRC Regulation of DOE Facilities." *Nuclear News*, p. 32. May 1997.

5.28 LOCATION OF THE REGULATOR

DOE believes the regulator should be located near the Hanford site. The NRC staff anticipates that NRC regulation of TWRS/WTP facilities would involve some staff FTEs (full time equivalents) at the Hanford site, some staff FTEs at the Region IV Office, and some staff FTEs at NRC Headquarters. The actual numbers would be determined by Congress via the enabling legislation and the scope of NRC regulation it requires.

5.29 MULTI-STEP LICENSING

The program for TWRS/WTP is split into several phases, and a multi-step licensing process may be more appropriate. The revised 10 CFR Part 70 allows multi-step licensing. An amendment process would allow the constructed TWRS/WTP facility the opportunity to obtain approval for changes, different flow capacities, and different radionuclide inventories. This is not an issue.

5.30 DIFFERENCES IN 10 CFR PART 20 AND 10 CFR PART 835

DOE believes differences in 10 CFR Part 20 and 10 CFR Part 835 need to be understood and reconciled to the extent necessary. For the most part, DOE and NRC radiation protection regulations are similar, bur a few significant differences exist. The most notable difference is that requirements found in 10 CFR Part 20 are more comprehensive that the 10 CFR Part 835 requirements, but where differences in dose values exist, the DOE requirements are usually more conservative. Both agencies also agreed that the key to developing a 10 CFR Part 20 compliant Radiation Protection Program (RPP) lies in the details of the implementation of the Safety Requirements Document (SRD) Safety Criteria for radiation protection. The TWRS-P Contractor had not developed the implementation details. It was expected that the level of detail needed to better evaluate compliance with 10 CFR Part 20 would become available at the PSAR stage when the TWRS-P Contractor would have submitted a revised RPP for construction. It is expected that a similar approach will be taken by the new contractors working under M&O arrangements.

The NRC staff does not consider this a transition issue.

5.31 COST OF CONTRACTOR DOCUMENTATION

DOE expects there may be some costs associated with changing documentation to meet NRC expectations after regulatory transition. Regulation by the NRC may require additional documentation, different formats, more specificity, or additional analyses. It would be expected that if the TWRS/WTP contractors were in compliance with the DOE regulatory framework at the time of transition to NRC regulation, the costs for revision of new plant documents (all of which would likely be available in electronic formats) would be small. Given the use of M&O contracts, DOE would fund these documentation requirements and changes.

The NRC staff does not consider this a significant transition issue.

5.32 THE ROLE OF DOE ORDERS VERSUS NRC RULES

DOE is concerned that there may be misunderstandings regarding the applicability and enforceability of DOE Orders and their relationship with DOE rules.

In the transition of regulatory oversight from DOE to the NRC for the GDPs, the NRC presumed that the GDPs were required to strictly comply with DOE Orders. However, that was not the case. Thus, the role of DOE Orders in the conduct of DOE business requires clarification. The term "Order" or more precisely, "Safety Order" is a misnomer and leads one to a basic misconception in understanding the DOE regulatory process. The terminology problems can be exacerbated because the NRC regulatory structure also includes the term "Order" but it is used in the more conventional sense. NRC Orders have the force of law and are issued to licensees as mandatory requirements. Contrary to what the term "Order" implies, DOE Safety Orders are not, on their own, legal requirements and are not necessarily mandatory.

Order-related lessons learned from the GDP transition are not generally applicable to TWRS. DOE Order compliance was specifically excluded from the TWRS-P contract.[94] Both the DOE and NRC have been involved in developing the TWRS-P regulatory (Integrated Safety Management Plan [ISMP] and SRD) and design (ISA and Safety Analysis Report [SAR]) bases. The regulatory basis for the new contracts has not been defined yet.

The NRC staff does not consider this an issue.

5.33 THE ROLE OF NATIONAL ENVIRONMENTAL POLICY ACT IN REGULATORY TRANSITION

DOE believes the potential effects of National Environmental Policy Act (NEPA) need to be considered for regulatory transition. Since the TWRS/WTP project is a major Federal action, DOE has prepared an environmental impact statement (EIS). The EIS considers the actual cleanup activities and operations of the TWRS-P facilities and includes its construction, operation, and decontamination and decommissioning activities.

The NRC staff does not consider this an issue because of the existing NEPA coverage and the similarities between the two agencies in their approaches to NEPA

5.34 ROLE OF THE ACRS OR ACNW IN REGULATORY OVERSIGHT TRANSITION

DOE is concerned by the potential effects of the Advisory Committee on Reactor Safeguards (ACRS) and/or the Advisory Committee on Nuclear Waste (ACNW) on the transition of TWRS/WTP regulation from DOE to NRC.

The ACNW would likely take an interest in the transition of regulatory responsibility associated with TWRS-P. A review of ACNW reports from July 1996 to the present suggests that a typical ACNW recommendation would be expected within 3 months of an initial presentation of the issue to the Committee. Some issues, that were not time sensitive, were not concluded for up

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[94] Department of Energy (U.S.)(DOE) Rev. 1 "Radiological, Nuclear, and Process Safety Regulation of TWRS Privatization Contractors," Regulatory Plan. DOE: Richland, Washington. January 1998.

to a year, while reports on other issues were released within 1 week of the presentation to the Committee. Records of recent meetings suggest that ACNW involvement in the TWRS-P regulatory oversight transition process would not result in any delays. ACNW participation in the process would occur in parallel with other transition/licensing activities.

The NRC staff does not consider this to be a significant issue because no delays are anticipated.

5.35 IMPACT ON FUTURE GENERATIONS

DOE is interested if there would be any impacts from regulatory transition upon future generations and resources. The premise for the TWRS-P vitrification plant is that it will take waste, process it, and return a stable vitrified waste form for stable, long-term storage and disposal. By doing so, DOE intends to remediate the Hanford waste tanks and, eventually, in the mid-21st Century,[95] return the land to beneficial us as it was prior to 1940. Future generations in the latter half of the 21st Century would benefit by greater land resources than are available now. External regulation of DOE facilities, including TWRS/WTP facilities, is expected to improve the safety and protection of workers, the public, and the environment. This can only have a positive effect upon the future.

The NRC staff does not see this as an issue for regulatory transition.

5.36 PUBLIC INPUT TO REGULATORY DECISIONS

DOE is concerned that regulatory decisions may not continue to be open to the public and public input after transition of regulation to the NRC. The transfer of TWRS-P to NRC regulatory oversight would result in the NRC licensing or certifying the TWRS-P Contractor's facility. In any licensing action, NRC is required to obtain public participation under 10 CFR Part 2, the Rules of Practice. In this process, the interaction is more formal and judicially governed than in the DOE case. NRC generally conducts interactions with the licensee in the public domain to the maximum extent possible. The ongoing pre-licensing activities on the mixed oxide project are a good example. The NRC staff does not consider this an issue as NRC practices and procedures usually result in more openness with the public than DOE regulatory actions.

[95] The Tri-Party Agreement specifies complete clean-up of tank wastes by 2028, following which time the tanks themselves and all appurtenances would require decontamination and decommissioning and disposal.

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6.0 FUTURE ITEMS, EVENTS, AND CONCERNS

The U.S. Department of Energy (DOE) has stated its intention to maintain the schedule as it transitions to the new contractor and Management and Operations (M&O) contracting arrangement. The original schedule at the time of this writing was (December 2000):

RPP:	Radiation Protection Program for Design, Construction, and Operation.
LCAR:	Limited Construction Authorization Request - due at the end of June 2000.
Seismic Issues:	Subject of the Topical Meeting, end of July 2000.
ISMP:	Integrated Safety Management Plan - revision due at the end of August 2000.
HAR:	Hazards Analysis Report - revision due at the end of September 2000.
SRD:	Standards Requirement Document - revision due at the end of October 2000.
CAR:	Construction Authorization Request - due early February 2001.

No revised schedule for these regulatory events is currently available.

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