COMPLIANCE EVALUATION REPORT

FOR THE CERTIFICATION OF THE

UNITED STATES ENRICHMENT CORPORATION

PORTSMOUTH GASEOUS DIFFUSION PLANT

PORTSMOUTH, OHIO

DOCKET 70-7002

AUGUST 1996

U. S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Division of Fuel Cycle Safety and Safeguards
Washington, DC 20555

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Chapter 1 INTRODUCTION

1.1 Introduction

This report documents the United States Nuclear Regulatory Commission (NRC) staff compliance evaluation of the U.S. Enrichment Corporation (USEC) certification application for the Portsmouth Gaseous Diffusion Plant (PORTS) located near Portsmouth, Ohio. The Portsmouth plant enriches uranium to a maximum of 10 percent U-235 by the gaseous diffusion method. The application consists of a safety analysis report (SAR), a quality assurance program, technical safety requirements (TSRs), emergency plan, environmental compliance status report, fundamental nuclear material control plan, transportation protection plan, physical protection plan, security plan for the protection of classified matter, waste management program, decommissioning funding program, environmental information, and a Department of Energy (DOE) prepared and approved compliance plan. USEC submitted its initial application for certification by letter dated April 18, 1995. By letter dated May 5, 1995, the NRC rejected USEC's application because it did not meet the requirements of Title 10 of the Code of Federal Regulations, Parts 76.31 and 76.35. The NRC concluded that the application did not contain adequate information for the NRC to independently determine that public health and safety, worker safety, the environment, and special nuclear material and classified matter would be adequately protected. The NRC and USEC staffs met numerous times to discuss the application contents. USEC submitted near-final sections for NRC advance review for acceptability. By letter dated September 15, 1995, USEC submitted a revised application. The Compliance Plan was submitted on November 7, 1995.

The staff requested additional information on the application by letters dated October 4, October 6, October 13, October 16, October 18, October 25, November 16, December 1, December 11, December 18, 1995, January 29, February 20, June 19, and July 9, 1996. USEC responded by letters dated October 25, November 8, November 9, November 16, November 17, November 22, December 1, December 8, December 12, December 13, December 14, December 22, 1995, and January 3, January 5, January 10, January 11, January 18, January 19, February 1, February 9, February 12, February 13, February 16, February 20, February 23, February 29, March 1, March 6, March 7, March 8, March 13, March 19, March 20, March 27, March 28, March 29, April 8, April 11, April 18, April 22, April 25, April 26, May 2, May 6, May 8, May 15, May 17, May 22, May 31, July 2, July 11, July 18, July 19, July 26, July 30, and August 1, 1996. The application was revised on January 19, May 31, July 26, August 1, and August 12, 1996.

The staff requested additional information on the Compliance Plan by letters dated December 15 and 18, 1995, February 20, 1996, and June 19, 1996. USEC responded by letters dated January 19, February 20, March 27, April 2, April 4, April 9, April 18, April 26, April 30, May 1, May 9, May 24, May 30, June 10, June 17, July 12, July 19, and August 1, 1996. USEC submitted revision 2 of the Compliance Plan on February 5, 1996. Revision 3, without 3 issues, was submitted on July 12, 1996; the remaining issues were submitted by letters dated July 15 and July 18, 1996. In addition to the correspondence, NRC, USEC, and DOE met many times to discuss application questions and had numerous conference calls.

The application and all non-proprietary, unclassified supporting information and communications are available at the NRC Public Document Room (2120 L Street, N.W., Washington DC 20555) and at a local public document room at the Portsmouth Public Library, 1220 Gaily Street, Portsmouth, Ohio 45662 under Docket 70-7002.

As part of the staff's consideration of the application, there was a public comment period on the application. Notice appeared in the Federal Register (60FR49026) on September 21, 1995, allowing for a 45-day public comment period on the application. Another notice appeared in the Federal Register (60FR57253) on November 14, 1995, providing for a 45-day public comment period on the Compliance Plan. In addition, a public meeting was held on December 28, 1995, at the Vern Riffe Joint Vocational School in Piketon, Ohio. The comments received at the meeting and during the comment period are discussed in Appendix A.

1.2 Background

The President signed H. R. 776, the "Energy Policy Act" (the Act), into law on October 24, 1992. Among other things, the Act amended the Atomic Energy Act of 1954 to establish a new government corporation, the U.S. Enrichment Corporation for the purpose of managing and operating the uranium enrichment enterprise owned and previously operated by DOE. The Act provided that within two years after enactment of the legislation, the NRC is to promulgate standards that will apply to the two operating gaseous diffusion plants to protect public health and safety from radiological hazards and to provide for the common defense and security.

The Act further directs the NRC to establish a certification process under which the two gaseous diffusion plants at Piketon, Ohio, and Paducah, Kentucky, will be operated by USEC and will be certified annually by the NRC for compliance with those standards. The Act also requires the NRC to report annually to Congress on the status of the gaseous diffusion plants.

On February 11, 1994, the Commission published in the Federal Register for comment a proposed new Part 76 to Chapter I of Title 10 of the Code of Federal Regulations which establishes requirements and procedures for the certification process. The final rule was published on September 23, 1994 and became effective on October 24, 1994. Under a Memorandum of Understanding between NRC and DOE, DOE is to continue nuclear safety, safeguards and security oversight of the gaseous diffusion plants until the NRC has certified that the facilities are in compliance with the standards (Part 76) or has approved a plan for achieving compliance with the standards.

DOE remains responsible for decommissioning of the site and retains ownership of the facilities. The Environmental Protection Agency (EPA) jurisdiction remains unchanged. The NRC consulted with EPA as part of the certification process. The Occupational Safety and Health Administration has regulatory responsibility for occupational safety at the facilities. Consultation with these other agencies is discussed in Appendix B. The USEC leased the two gaseous diffusion plants from DOE beginning July 1, 1993. The plants have operated continuously since the early 1950s.

The objective of this review is to determine whether PORTS operations comply with the regulations contained in 10 CFR Part 76. The review considers the management organization and administrative programs provided to assure safe operation of the facility. The review identified and evaluated those elements of plant operation, termed important to safety, which must function at the highest level of reliability.

1.3 General Plant Description

The regulations at 10 CFR §76.55(a)(1) require USEC to include in the SAR the "activities and locations involving special nuclear material and the general plan for carrying out these activities." This information is provided in Chapters 1, 2, and 3 of the SAR. The regulations at 10 CFR §76.35(a)(2) also require USEC to provide the "name, amount, and specifications (including the chemical and physical form and, where applicable, isotopic content) of the special nuclear material, source and byproduct material the Corporation proposes to use, possess or produce, including any material held up in equipment from previous operations."

1.3.1 Site Description

PORTS is located on an approximately 3,708-acre Federally-owned reservation in Pike County, Ohio. The site is in a generally rural area that was previously farmland and the watershed for several intermittent streams. The largest cities within a 50-mile radius are Portsmouth, Ohio (from which the plant takes its name), located approximately 27 miles to the south and Chillicothe, Ohio, located approximately 27 miles to the north. Portions of 24 counties are located within a 50-mile radius of the plant, 18 of which are in Ohio, 5 in Kentucky, and 1 in West Virginia.

1.3.2 Plant Description

1.3.2.1 Overview

Using the gaseous diffusion process, PORTS produces enriched uranium for use in fuel assemblies for nuclear power reactors. Manufacture of reactor fuel involves several distinct steps, each of which is carried out at a different plant and at a different location. The first step is the mining and milling of uranium ores to produce yellow cake. The yellow cake is then converted into uranium hexafluoride ($UF_{6)}$. Next, the UF_6 is shipped to an enrichment facility, such as PORTS, where the concentration of fissionable ^{235}U is increased. Finally, the enriched UF_6 is transported to a fuel plant where it is processed and fabricated into fuel assemblies for reactors.

Enrichment through gaseous diffusion depends on the separation effect arising from molecular effusion (i.e., the flow of gas through small orifices). When a mixture of gas molecules is confined in a vessel, the average velocity of the lighter molecules is greater than that of the heavier molecules. Therefore, the molecules of the lighter gas strike the vessel walls more frequently than the molecules of the heavier gas. If the walls of the vessel are porous with holes large enough to permit escape of individual molecules, but sufficiently small so that bulk flow of the gas is prevented, then the lighter gas molecules

escape more readily than the heavier ones. The gas consisting of the escaped molecules is thus enriched in the lighter components of the mixture.

The main components for carrying out the enrichment process at PORTS are: large cylindrical vessels called converters that contain porous barrier tubes; compressors used to compress the UF_6 gas to pressures needed for flow though the barrier tubes; electric motors to drive the compressors; heat exchanger and a cooling circuit for removing the heat of compression; and piping for stage and interstage connections with control valves to adjust gas flow. In addition to this process equipment, there are UF_6 feed and withdrawal systems, an extensive electrical power distribution system, and cooling towers to dissipate the waste process heat.

The basic unit of the gaseous diffusion process is the converter, which is the vessel in which the diffusion takes place. A long series of converters is needed. Compressed UF_6 gas is made to flow inside an array of porous barrier tubes inside each converter. Approximately one-half of the gas passes through the barrier tube pores into a region of lower pressure. This gas is enriched in the component of lower molecular weight (^{235}U) and is routed to the next converter for further enrichment. The gas that does not pass through the barrier pores is depleted with respect to ^{235}U and is routed to the previous converter. Upon leaving the converter chamber, the enriched and depleted streams have to be recompressed to the barrier tube high-side pressure to make up for frictional losses.

The degree of enrichment achieved in a single converter stage is very small. To achieve useful enrichment levels, the effect must be multiplied many times over by making use of a cascade of numerous converter-compressor stages in series. The number of stages required is determined by the enrichment needed. PORTS has 2400 stages currently active. To facilitate maintenance activities and to minimize resulting productivity losses, stages are grouped together to form cells. Each cell has inlet and outlet block valves enabling it to be taken offstream, bypassed, and shut down. A cell is the smallest unit of equipment that can be isolated from the cascade for maintenance service. Each cell consists of a number of converters and the supporting equipment, primarily the compressors and heat exchangers. For practical operation cells are grouped together to form units. Each unit consists of cells which are of the same equipment size and which usually operate under the same conditions. The location of any particular converter can be identified by specifying the equipment size, unit, cell, and stage. For example, the converter in unit three, cell five, stage one in the size X-33 equipment would be specified by X-33-3-5.1.

The capacity of a plant to enrich uranium is expressed in separative work units (SWUs). PORTS is rated at 8.6 million SWUs annually, with a power consumption of 2260 megawatts of electricity.

In an ideal cascade, each stage would be slightly different from the stages immediately above or immediately below. The converters that contain the largest barrier area would be located at the normal assay (0.711% ^{235}U) feed point. Feed refers to the UF_6 that is injected into the cascade for enrichment. Stages above the feed point would be progressively smaller and are referred to collectively as the enriching section. The stages

below the feed point would also be progressively smaller and are referred to as the stripping section.

The number of stages required in each section is determined by projected operating parameters and by the total enrichment and depletion planned. PORTS has the capability of producing reactor grade (currently up to 5.0% ^{235}U with a design capability to produce up to 10% ^{235}U) product. For typical operation at PORTS, there are about 1120 stages between the 0.711% feed point and the point where 4.9% product is withdrawn. These stages comprise the "enriching" section of the cascade. The number of stages in the enriching section can be varied significantly through selection of the withdrawal point to accommodate operating conditions. Additional stages are used to strip the ^{235}U isotope from normal feed material so that the depleted material or tails will contain about 0.2 to 0.3% ^{235}U (this depends upon power load and marketing considerations). The stripping section is made up of about 620 stages. An additional 180 stages, referred to as purge stages, separate and purge the light gases that leak into, or are deliberately introduced into, process equipment. Between the product withdrawal point and the purge stages there are about 480 additional stages which serve as a buffers. Thus, as currently configured, the Portsmouth plant comprises about 2400 stages.

Economics dictates the use of large blocks or groups of process equipment that have components of the same size and operating characteristics, thus limiting the number of different basic sizes. At PORTS, five basic separative equipment sizes (compressors and converters) are used in the isotopic stages. Nuclear safety requires smaller equipment and piping sizes as the ^{235}U assay increases in the cascade. The five sizes, from largest to smallest, are designated:

a. 000 or X-33 size (largest).
b. 00 or X-31 size.
c. 0 or X-29 size.
d. X-27 size.
e. X-25 size (smallest).

1.3.2.2 Process Buildings and Operations

The following contains a brief description of the operations that occur in each of the main buildings.

Process equipment is housed in three steel-framed, transite-covered, two-story buildings designated X-333, X-330 and X-326. In each of these buildings the equipment is on two floors, with the diffusion equipment being on the cell (upper) floor and the electrical switchgear and control instrumentation being on the operating (ground) floor.

The X-333 Process Building is 1456 feet long, 970 feet wide, and 82 feet high. It's two floors have combined floor space of approximately 65 acres. It houses 640 stages (organized into 80 cells, 8 units) of X-33 (or 000) size equipment, which is the largest size equipment. This array of equipment is connected into the cascade flow between units 2 and 3 of X-31 (00) size equipment, located in the X-330 Building.

The X-330 Process Building is 2176 ft long, 640 ft wide, and 66 ft high. Its two floors have a combined floor space of approximately 55 acres. It houses 500 stages (50 cells, 5 units) of X-31 size equipment, and 600 stages (60 cells, 6 units) of X-29 (0) size equipment. Units X-31-2, X-31-1 and X-29-1 are in the stripping section of the cascade and are below the X-33 units in the cascade flow arrangement. The depleted stream from Cell 10 of X-29-l becomes the "tails" of the process. This material is withdrawn at a special facility located at the northeast corner of the building. The remaining units progressively enrich the flow until it exits the building en route to Building X-326.

The X-326 Process Building is 2230 ft long, 552 ft wide, and 62 ft high. Its two floors have a combined floor space of approximately 58 acres. It houses 720 stages of X-27 equipment, 1440 stages of X-25 equipment, and 180 stages of purge cascade equipment. Of the X-25 and X-27 equipment, 1680 stages have been retired in place; they were associated with former high enriched operations under DOE but are not used in the low enriched operations covered by this certification. Cascade flow from the X-330 Building enters the building at the northeast corner and is directed into the first cell in Unit X-27-1.

1.3.2.3 Cascade Shape and Tapering

Several factors influence cascade "shape." A cascade producing at a high product rate must have higher flow rates between stages than one producing at a lower product rate. To utilize power efficiently, the stages in the middle (or near the feed point) must have higher flow rates than those at the ends. This is usually represented or drawn in a diamond shape to illustrate this flow distribution. The PORTS cascade is tapered by appropriate sequencing of the five different equipment sizes and by reducing the pressures across each size equipment.

The middle part of the cascade, where feed is introduced, contains the largest, highest pressure cells resulting in the maximum interstage flow rate near the feed point. The flow taper is achieved in units of the same equipment size by gradually decreasing the pressure from cell to cell and unit to unit. At approximately 320 stages away from the feed point, the equipment size is to reduced (to X-31 size) and the pressure is raised to compensate for the smaller equipment. Pressure tapering is then resumed as the equipment size is further reduced; pressures would again be tapered down toward the cascade ends in order to achieve efficient flow distribution.

The product withdrawals from the PORTS cascade are usually made up of two different streams. Most product withdrawals take place at two special stations. The cascade feed is made up of two (sometimes three, including high enriched uranium (HEU) refeed streams of different assays. The lower feed stream is usually normal assay material; the higher feed consists of slightly enriched product from the Paducah Gaseous Diffusion Plant. At every point where there is a feed or withdrawal, the flow taper required for the most efficient power utilization and assay control for withdrawal operations must take into account the sudden change in cascade flow conditions. In general, a different taper is required for each feed and withdrawal location. The number of cells may vary as the lower product assay changes. Normally, cells are started and shut down and the cascade retapered to maintain high efficiency when the lower product assay is changed or as power availability dictates.

1.3.2.4 Feed and Sampling

Sampling is carried out to allow UF_6 purity measurements and assay (ratio of ^{235}U mass to total U mass) measurements. High purity is essential to ensure uniform, predictable behavior of reactor fuel. Both purity and assay are measured at many locations in the enrichment process, including those for feed and withdrawal. Gas samples are simpler to obtain and are acceptable for assay measurements, but liquid samples give more accurate results for purity measurements. The availability of liquid UF_6 at feed and withdrawal locations accounts for the close association of sampling with those locations.

The following facilities perform feed and sampling functions:

- X-343 Feed Vaporization and Sampling Facility,
- X-342A Feed Vaporization and F2 Generation Facility,
- X-344A Toll Enrichment Services Facility,

The major cascade feed streams are supplied through feed headers connected to the feed autoclaves in the X-342 and X-343 Feed Facility. The X-343 Facility, which has a total floor area of 11,200 square feet, is the receiving point for all inbound UF_6 natural assay (0.711% ^{235}U) and Paducah Product (up to 2.75% assay ^{235}U) feed material. This feed material is received in large cylinders (10-ton and 14-ton). The amount of feed material on hand will vary, but typically there will be a 90-day supply of feed in the X-343 cylinder storage areas. The X-343 Facility is also the shipping point for the empty cylinders after they are fed to the cascade.

The X-343 Facility is equipped with seven steam-heated autoclaves. All seven are designed for feed or vapor-phase sampling operations and are connected to the cascade feed headers. Of the seven autoclaves, three have seven-foot diameters and are equipped with cylinder rollers to allow the cylinder valve to be positioned below the UF_6 liquid/gas phase boundary for liquid phase sampling. The remaining four have six-foot diameters and do not have roller capability.

In a typical feed operation, a 10-ton cylinder of UF_6 at ambient temperature is lifted by crane and placed inside an open autoclave. The cylinder valve is then connected by a flexible tube to a cascade header. Various purges, safety tests, and procedures are carried out, after which the autoclave is closed. Steam is turned on and after a matter of hours, the UF6, which was a soft solid at ambient temperatures, liquifies with an attendant rise in the vapor pressure of UF_6 gas in the cylinder volume above the liquid surface. Header valves are now opened, exposing the UF_6 gas to the suction pressure of the cascade. The gas is drawn into the cascade and the enrichment process begins. The mass of UF_6 in the cylinder progressively diminishes with time and within a few hours, typically 9 or 10, the mass is reduced from the initial 10 metric tons to a few kilograms. The autoclave is valved out and a prepared cylinder in one of the remaining six autoclaves is valved in to take over the feed function. After cooling, the original autoclave is opened, the cylinder tubing is purged and disconnected, and the almost empty cylinder is lifted out by crane and prepared for shipment back to the UF_6 supplier.

The X-342A Facility was the former primary location for feed sampling and cascade feed vaporization. The facility is equipped with two feed-and-sample autoclaves and serves largely as a back-up feed station.

In addition to the principal feed streams discussed above, a high enriched uranium (HEU) stream is fed to the cascade. This operation is carried out to make use of excess HEU no longer needed for government purposes. The operation is not a necessary or permanent part of PORTS operations. It consists of varying assays of DOE material that is blended down to commercial assay requirements. This feed is supplied through auxiliary headers connecting the cascade with condensation facilities in the Product Withdrawal (PW) area located in the southwest corner of the X-326 Building. The feed cylinders are small, each containing only a few tens of kilograms of UF_6. The feed process depends on sublimation. No autoclaves and little or no cylinder heating is involved.

1.3.2.5 Withdrawal of Tails and Product

Three permanently established facilities are provided for the withdrawal of intermediate assay UF_6 (1.0% to 5% ^{235}U) and the depleted UF_6, which is often referred to as "tails" (0.2% to 0.47% ^{235}U). The three are designated as: (1) the Tails Withdrawal Facility, which is located in the northeast corner of the X-330 Process Building; (2) the Extended Range Product (ERP) Withdrawal Station, which is in the northeast corner of the X-326 Process Building; and (3) the Low-Assay Withdrawal (LAW) Station is in the west center of the X-333 Process Building.

Uranium hexafluoride is withdrawn in liquid form at these three locations. To accomplish this, the gas stream is compressed to a pressure and temperature above the triple point and then cooled to condense the vapor. The liquid condensate is then drained into UF_6 cylinders which are 30-inches or 48-inches in diameter, depending on assay. The compressors and their gas coolers and condensers are located on the second floor. The accumulators, associated piping, UF_6 condensers, and valves are located on the mezzanine level between floors. Withdrawal stations (manifolds, scales and carts) are located on the ground floor of the process building. The compressors, coolers, accumulators, manifolds and associated piping are enclosed in heated housings which are maintained at approximately 200°F.

Depleted UF_6 is withdrawn at the Tails Withdrawal Facility, but in the case of its shutdown, the ERP and LAW Stations could be used to perform this function. Each product withdrawal facility has the capability of performing two withdrawals simultaneously at different assays.

1.3.2.6 Shipping and Receiving

The X-344A Toll Enrichment Services Facility is the processing and shipping/receiving point for low assay (5% enrichment or less) toll product UF_6. Toll services allow a customer, for a fee, to supply uranium on one enrichment, usually natural, and receive UF6 of a higher enrichment. The building has approximately 40,000 square feet of ground floor area, 18,700 square feet of second floor, and 4,300 square feet of basement floor area. The majority of the ground floor comprises two high-bay areas which are served by overhead

cranes and have large roll-up doors to allow entry of semi-tractor/trailer rigs, straddle carriers, forklifts, and railcars. Cool down positions are accessed by crane through the west side of the building.

Typically, toll product is withdrawn from the enrichment cascade into 10-ton government-owned cylinders. Before the material can be shipped, it must be sampled and transferred into the smaller (2 1/2-ton) cylinders which are usually owned by the customers and are approved for transport over highways and railways. The sampling and transfer operations are carried out in X-344A using any of the four autoclaves present in the north high-bay. Special shipping packages are used to protect the full product cylinders in shipment.

Shipping and receiving activities can be carried out in either the north or south bay using any of the overhead bridge cranes. Cylinder unloading can occur outside the building in the X-745-B cylinder storage lot during good weather.

1.4 History of Gaseous Diffusion Plant (GDP) Operation

Design studies for the plant began in June 1951 when the Atomic Energy Commission (AEC) determined that another gaseous diffusion plant should be added to the Oak Ridge and Paducah complex to provide ^{235}U production at rates substantially above those of the existing complex.

In August 1952, the Commission announced a site selection after exploration of many possible locations. Its choice was a 4000-acre tract in the Ohio River Valley along the Scioto River in Pike County, Ohio, which met site selection criteria including the availability of a vast expanse of relatively flat terrain, a reliable supply of large amounts of low-cost electrical power, a dependable source of water, local labor, and suitable transportation networks. Construction of the plant began in late 1952.

In September 1952, the Goodyear Tire & Rubber Company was named management contractor. Goodyear Atomic Corporation was established as a wholly owned subsidiary of the Akron-based company to be responsible for managing the government plant. Management of the total DOE uranium enrichment complex became the responsibility of Martin Marietta Energy Systems, Inc. in November 1986 with the transfer of the contract from Goodyear.

The first stage of gaseous diffusion separative equipment began operating in September 1954, and the first full production cell came "on line" in January 1955. In March 1956, the plant was completed at a cost of $750 million. Peak construction employment was 22,500 in the summer of 1954. Contractors cleared more than 1200 acres and moved more than 4.5 million cubic yards of earth during construction.

Originally built for and dedicated to national defense missions, PORTS began serving the commercial nuclear power industry in the 1960s. As the nuclear power industry grew during the 1960s, electric utilities gradually became the principal customers of uranium enrichment services. Electric utilities worldwide purchase up to five percent ^{235}U for fuel in nuclear power reactors.

The plant has undergone several major improvement programs. From 1974 to 1982, the combined Cascade Improvement Program/Cascade Uprating Program (CIP/CUP) was accomplished, incorporating into the plant the best available technology for gaseous diffusion uranium separation.

PORTS has not produced HEU directly for nuclear weapons use since 1964. It has continued to produce HEU of more than 95 percent ^{235}U for the U.S. Navy's fleet of nuclear-powered ships (submarines and aircraft carriers). In November 1991, the Secretary of Energy announced the suspension of HEU production, and that portion of the process system is now being shut down. This activity is scheduled to be completed by November 1994. Portions of the X-326 Building, which was used for the production of HEU, will continue to be used for commercial-level enrichment.

All shipments of enriched uranium to DOE customers have been made from PORTS since 1985 when the Oak Ridge diffusion plant was shut down. Most plant product is enriched to between two and five percent ^{235}U and shipped to fuel assembly fabricators prior to use in electric utility nuclear reactors.

1.5 Site Operations Summary

In the United States, three production size gaseous diffusion plants have been built: one at Oak Ridge, Tennessee; one at Paducah, Kentucky; and the subject plant at Portsmouth, Ohio. Collectively, the three plants have over 100 years of operating experience. During that time, no incidents at any of the GDPs have caused death or serious injuries to plant personnel from exposure to radioactive materials or radiation nor have there been any incidents that have resulted in off-site releases of radiation or radioactive materials that could cause committed doses in excess of established limits (DOE 7/1/93 safety basis document). Nonetheless, there have been UF_6 releases from time to time at each of the plants. A summary of the six (six was chosen arbitrarily) largest releases of UF_6 at PORTS is given in the table below.

SUMMARY OF THE SIX LARGEST UF_6 RELEASES AT PORTS

Location	Date	Description	KG U
Liquid Cylinder Storage	3/78	Cylinder rupture from drop during transport	5926
Tails Withdrawal	10/78	Cylinder valve broke as result of transport while connected	561
Tails Withdrawal	7/69	Cylinder valve would not close	460
Transfer Bay	9/76	Pigtail connection leak	65
Sampling	5/73	Cylinder valve would not close	45
Feed Vaporization	7/65	Pigtail rupture	14

The CIP/CUP, which were completed in 1984, incorporated appropriate safety improvements based, in part, upon the GDP operating experience gained since the 1950s. The GDPs produced a Final Safety Analysis Report (SAR 85) for each site between 1980 and 1985. These documents identified the major safety events (major hazards, initiators,

and sequences) and established an envelope for safe operation, as defined in DOE orders. The SAR 85s analyzed the unique risks associated with the operation of the GDPs, examined the impact of these hazards during accident and process upset conditions, and evaluated the risk to both on- and off-site personnel. According to DOE, the SAR 85 development utilized operational experience, engineering judgement, an in-depth understanding of the overall operation and associated hazards, and senior staff with first-hand experience and understanding of the theory and operations. In 1989, DOE initiated an upgrade program for the SAR 85s; this effort continues and is expected to be complete in February 1997. The upgrade program is discussed in greater detail in §15.3.

1.6 Authorized Activities

A complete list of PORTS authorized activities for each regulated material is provided in Table 1.4 in the application. With respect to unenriched uranium the authorized activities are:

1. Enrichment of uranium up to 10 percent enrichment by weight $_{235}$U.

2. Receipt, storage, inspection, and acceptance sampling of cylinders containing uranium.

3. Filling and storage of cylinders of natural uranium and uranium depleted in $_{235}$U.

4. Cleaning and inspection of cylinders used for the storage and transport of process product and tails containing source or spacial nuclear material (SNM).

5. Storage of process wastes containing uranium, transuranic elements, and other contaminants and decay.

6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes.

7. Radiation protection, process control and environmental sample collection, analysis, instrument calibration, and operation checks.

8. Maintenance, repair, and replacement of process equipment.

9. Laboratory analysis and testing.

10. Heating cylinders and feeding contents into the diffusion process.

11. Controlled feeding of cylinders.

12. Transfer between cylinders.

With respect to enriched uranium, the authorized activities are:

1. Filling, assay, storage, and shipment of cylinders containing uranium enriched up to 10 percent by weight $_{235}U$.

2. Nondestructive testing and analyses of product and process. streams.

3. Receipt, storage, inspection, and acceptance sampling of cylinders containing uranium enriched up to 10 percent by weight $_{235}U$.

4. Cleaning and inspection of cylinders used for the storage and transport of process feed, product, and tails containing source or SNM.

5. Storage of process wastes containing uranium, transuranic elements, and other contaminants and decay products.

6. Process, characterize, package, ship, or store low-level radioactive and mixed wastes.

7. Radiation protection, process control and environmental sample collection, analysis, instrument calibration, and operation checks.

8. Maintenance, repair, and replacement of process equipment.

9. Laboratory analysis and testing.

10. Heating cylinders and feeding contents into the diffusion process.

11. Controlled feeding of cylinders.

12. Transfer between cylinders.

13. Uranium remaining in equipment and facilities as a result of previous operations.

Except for Paducah product and the "stockpile" UF_6 transferred from DOE to USEC for enrichment, uranium to be fed to the cascade will meet the requirements of American Society for Testing and Materials (ASTM) Standard C996, "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% ^{235}U" or ASTM Standard C787, "Standard Specification for Uranium Hexafluoride for Enrichment" for reprocessed UF_6. All other uranium that does not meet the requirements of ASTM C996 or C787 for reprocessed UF_6 may be accepted for storage and subsequent dispositioning but will not be introduced to the cascade, with the exception of small amounts (e.g., 50 pounds UF_6) associated with sampling, subsampling, and analyses required to establish receivers's values.

The activities listed In Table 1.4 in the application are those which the NRC has reviewed and will certify that they meet regulatory requirements. If additional activities are planned, USEC will need to perform a safety analysis and propose TSRs as necessary prior to conducting the activity.

1.7 Codes and Standards

Appendix A to Chapter 1 of the SAR contains a list of the various industry codes and standards and NRC regulatory guidance documents that have been referenced in the PORTS application and in the responses to questions in the application. In many cases, the extent of the commitment needs to be clarified. Compliance Plan Issue 41 commits USEC to review the commitments and to compile a listing of the specific sections of codes, standards, and regulatory guides to which USEC is committed. The results of this review will be transmitted to the NRC within 90 days after certification. This will be complete prior to the NRC assuming jurisdiction. The staff concludes that the plan and schedule are acceptable.

Chapter 2 SITE CHARACTERISTICS

2.1 Geography and Demography

PORTS is located at latitude 39° 0' 30" N and longitude 83° 0' 00" W on an approximately 3,708-acre Federally owned reservation in Pike County, Ohio. The site is in a generally rural area that was previously farmland and the watershed for several intermittent streams. The largest cities within a 50-mile radius are Portsmouth, Ohio, located approximately 27 miles to the south and Chillicothe, Ohio, located approximately 27 miles to the north. Portions of 24 counties are located with a 50-mile radius of the plant, 18 of which are in Ohio, 5 in Kentucky, and 1 in West Virginia.

PORTS occupies approximately 700 security-fenced acres about 1.5 miles east of U.S. Route 23 and 2 miles south of State Route 32, and 2 miles east of the Scioto River.

2.2 Nearby Industrial, Transportation, Military, and DOE Facilities

Economic activity in the vicinity of PORTS consists primarily of farming, lumbering, and small businesses. In addition, a gravel quarry is located west of PORTS, adjacent to the Scioto River. The quarrying is done by surface extraction; no explosives are used.

The only significant industry in the vicinity is located in an industrial pa.k about 7 miles to the north. The industries include a cabinet manufacturer (2000 employees) and an automotive manufacturer (200 employees). It is not expected that these activities would have any impact on PORTS operations.

The primary roadways near PORTS are U.S. Highway 23 and State Highway 335, which traverse a roughly north-south course, and State Highway 124 (same as State Highway 32), which traverses an east-west course just north of PORTS.

Rail transportation in the area is provided by the N&S Railway and the CSX Railway.

The Pike County Airport is located approximately 11 miles north-northeast of PORTS. No commercial flights or cargo shipping occurs there. The 4,900-ft runway supports single and twin engine planes and small jets. The Greater Portsmouth Regional Airport, located approximately 15 miles southeast of PORTS, provides only light plane service (Class I airport). The nearest commercial airports are Port Columbus International Airport in Columbus, Ohio, approximately 70 miles away, and the airport at Huntington, West Virginia (87 miles away).

The Ohio National Guard maintains an area on the Portsmouth site for the reconditioning and storage of equipment. This equipment primarily consists of mobile equipment that contains no armament; no ordnance is permitted at this location. The maintenance and reconditioning of this equipment is accomplished in and around the X- 751 facility, located on the south end of the site. The storage of the equipment is primarily accomplished in an open field south of X-3001 and X-3002 (PB1 & PB2 of the Gaseous Centrifuge Enrichment Project (GCEP) site).

Although PORTS once maintained a landing strip for air transportation, the strip is now obstructed with earthen berms. The southern end of the landing strip is maintained as a helicopter pad. The shift supervisor coordinates helicopter approaches to ensure they do not fly over process buildings or hazardous material storage areas.

In addition to administrating the lease agreement with USEC, DOE conducts various operations on the reservation including environmental restoration, decontamination and decommissioning (D&D), remedial activities [Resource Conservation and Recovery Act (RCRA)]; waste management, treatment, storage, and shipment of low-level radioactive waste (LLRW) and mixed waste; manages DOE non-leased facilities; and is responsible for regulating highly enriched uranium.

2.3 Climatology and Meteorology

The PORTS environs have a generally moderate climate. Winters are characterized as moderately cold, having an average of 112 days when the low temperature is at or below 32°F and only 3 days when the temperature drops to 0°F or lower. Summers are moderately warm and humid, having an average of 27 days when temperatures reach or exceed 90°F.

Annual precipitation is approximately 40 inches per year and is distributed fairly evenly over the four seasons, fall being the driest season. Frequent passage of frontal systems results in the precipitation usually being well distributed over the area. Average annual snowfall is approximately 20 inches.

Prevailing winds at the site are out of the southwest to south. Average wind speeds are about 5 miles per hour (mph). Daytime wind stabilities are most commonly class D (neutral) or class C (slightly unstable). Nighttime winds are predominately class F (moderately stable).

Tornadoes are formed during warm, humid, unstable weather, and are usually associated with larger scale severe thunderstorm activity. Tornado wind speeds may reach 200 mph or greater, with their forward speed averaging 40 mph. Their path averages 1/4-mile wide and less than 16 miles in length, but some have been known to have paths over 1 mile wide and 300 miles long. For the period from 1953 to 1972, Ohio had 235 reported tornadoes. About 70% of them occurred between April 1 and July 31, and nearly 75% of them touched down between 2 p.m. and 10 p.m. Pike County had two tornadoes during the same 20-year period. The surrounding counties of Ross, Highland, Adams, Scioto, and Jackson and 3, 7, 2, 4 and 0 tornadoes, respectively. The accepted method for determining tornado probability consists of dividing the geographical area of the United States into one-degree-square regions. A tornado frequency, depending on the number of tornadoes occurring within the square during a specified length of time, is assigned to each one-degree-square. Using a variation of this method, the probability for any given year that a point within the 6-county area will lie within the damage path area of a tornado was calculated as 0.0008. This corresponds to a recurrence interval of over 1,000 years.

2.4 Hydrology

PORTS is located near the southern end of the Scioto River basin, which has a drainage area of 6,517 square miles. The headwaters of the Scioto River form in Auglaize County in north central Ohio. The river flows 235 miles through nine counties in Ohio, and through the cities of Columbus, Circleville, Chillicothe, and Portsmouth. At Portsmouth, in Scioto County, the river empties into the Ohio River at river mile (RM) 356.5. The slope of the Scioto River channel averages about 1.7 ft/mile between Columbus and Portsmouth. Five of the river's tributaries have drainage areas of over 500 square miles each: Olentangy River, Big Walnut Creek, Darby Creek, Paint Creek, and Salt Creek.

Water in the vicinity of PORTS is available from Lake White, the Scioto River, and groundwater supplies. Water used at PORTS normally comes from groundwater. Of an average 19 cubic feet per second (cfs) is for cooling water makeup and 4 cfs for sanitary purposes. Currently, all water is supplied by wells in the Scioto River alluvium. These wells are located near the east bank of the Scioto River, downstream from Piketon. Four well fields (X-605G, X-608A, X-608B, and X-6609) have the capacity to supply reliably between 36.4 and 40.2 cfs. PORTS can use water from the Ohio River through an intake at the Ohio River at RM 350.8, which is 5.7 miles upstream from the mouth of the Scioto River.

PORTS is generally free from problems of flooding. The source of the flood waters would be the Scioto River. The stage of the historic 1937 flood on the Scioto was 593.7 ft above mean sea level (MSL) at the Higby gauging station.

The corresponding flood stage of the Scioto River next to PORTS was estimated to be 556.7 ft by using the estimate that the Scioto River drops approximately 37 ft between the Higby gauging station (RM 55.5) and the mouth of Big Beaver Creek. PORTS has a nominal elevation of about 670 ft above MSL, which would be about 113 ft above the historical flood level for the Scioto River in the area.

Although PORTS elevation is greater than the maximum historic levels recorded for the Scioto River in the area or the 500-year flood levels predicted by the U.S. Army Corps of Engineers, a calculation of the "probable maximum flood" (PMF) was also performed. The details of a method of calculating the PMF are discussed in NRC Regulatory Guide 1.59, (NRC 1977, App. B). It is based on the drainage area and the location of the watershed involved. The drainage area of the Scioto River basin above Higby is 5131 square miles and that of the whole basin is 6517 square miles. The drainage area of the Scioto River above PORTS (RM 27.5) is between those two values. A conservative estimate for the PMF discharge of the Scioto River at either Higby or PORTS is approximately 1,000,000 cfs. This value is used as the PMF discharge of the Scioto River at PORTS, which including the wind/wave activity contribution, would correspond to a flood level of 571 ft., well below the nominal 670 ft. elevation of PORTS.

Calculations using two widely accepted probabilistic methods, the log Pearson III distribution and the Gumbel method, were carried out. The 10,000-year flood discharges of the Scioto River at Higby determined with these two methods are 526,000 and 280,000 cfs, respectively. Both of these discharge rates a smaller than that of the PMF. The

PMF is, therefore, the bounding event in determining the evaluation basis loads from flooding for PORTS.

Conservative estimates indicate that the failure of upstream dams would not threaten the safety at PORTS because of the high nominal plant grade elevation. In addition, the limited storage capacities of the reservoirs, the large stream distances of these dams from PORTS, and friction and form losses would make the actual wave heights even smaller than the estimated values. Discharges were considered of dam failures at full pool combined with that of either a 25-year flood or one-half of the PMF of the Scioto River. The result involving one-half of the PMF would result in a higher value, which is also somewhat greater than that of the PMF. However, this combined extreme flood would not threaten the safe operation of PORTS because of the high nominal plant grade elevation, similar to the case of the PMF.

2.5 Geology and Seismology

Geologically, PORTS is located within the Appalachian Plateaus physiographic province, about 20 miles south of its northwestern edge. It is bordered on the north and west by the Central Lowlands province, to the south and west by the Interior Low Plateaus province, and on the south and east by the Valley and Ridge province.

The Appalachian Plateaus province is composed of mature upland areas that have been dissected by erosion and now exhibit moderate to strong relief. This province is underlain by gently dipping Mississippian, Pennsylvanian, and Permian Age shale and sandstone. Both the adjacent Central Lowlands and the Interior Low Plateaus provinces are underlain by relatively flat-lying Paleozoic Age limestone and shale. The Interior Low Plateaus (Lexington Plain section) to the south and west is a mature to old plain of low relief; whereas, the Central Lowlands (Till Plains section) to the north and west is a young feature of low relief. The Valley and Ridge (Tennessee section) is underlain by thrust-faulted Paleozoic limestone, dolostone, shale, and sandstone of moderate relief.

Portions of the Appalachian Plateaus, Central Lowlands, and Interior Low Plateaus provinces have been glaciated, but the site is south of the region covered by Pleistocene glaciation. However, alluvium and transported glacial sediments form a surface veneer in the mile-wide, broad valley where PORTS is located. The surrounding hills have been maturely dissected by erosion, exposing the underlying, nearly flat-lying shale and sandstone of Mississippian and Pennsylvanian Age. Ground elevations within the plant generally range from about 660 ft MSL to 680 ft MSL, although the ground rises to about 700 ft MSL at the base of hills that border the Perimeter Road; the surrounding hills extend up to about 1,200 ft MSL.

Between 1776 and August 17, 1990, 264 earthquakes have occurred within 200 miles of the site. The location of the epicenters of the largest recorded earthquakes within 200 miles of the plant have been mapped. Taken into account are all tremors with magnitudes of 4.0 or greater, as well as all earthquakes where a magnitude has not been assigned. Events with a known magnitude of 3.9 or less were omitted from consideration. Two earthquakes of magnitude 5.0 or greater have occurred within this 200-mile radius in the 204-year period. The 1980 event had an epicenter in the central stable portion (Interior

Low Plateaus physiographic province) and the 1897 event occurred in the deformed Appalachian Highlands (Valley and Ridge physiographic province). The magnitude 5.1 event in northern Kentucky at 38.2°N, 83.9°W may be near the intersection of the Kankakee and Cincinnati Arches of Paleozoic Age, but the magnitude 5.80 May 31, 1897, event at 37.3°N, 80.7°W on the Virginia-West Virginia border is within the deformed Appalachian Highlands but apparently unrelated to any known tectonic feature. PORTS operating personnel indicate that the facility has performed without seismic damage or interruption of operations during its existence, and there have been no observed ground ruptures, sand boils, or subsidence at the site.

Including multiple fault systems and groups of related faults, 376 faults have been mapped within a 200 mile radius of PORTS. These have been compiled from existing published and unpublished geologic literature. Fault studies by the Tennessee Valley Authority (TVA), which contained information on 375 of these faults, show that only the "White Mountain Fault Zone" may be capable, i.e. exhibited movement at or near the surface in the past 35,000 years or movement of a recurring nature in the past 500,000 years. This fault is 20.5 miles in length and is located in Bell and Knox Counties, Kentucky, about 155 miles south-southwest of the site.

Five studies have been conducted specifically for determining the seismic hazard at PORTS. These studies were conducted by Blume, Dames and Moore, TERA, Wiggins, and Law Engineering. The first three studies were directed toward gaseous diffusion enrichment buildings while the last two were directed toward the gas centrifuge enrichment plant buildings. The PORTS SAR, submitted as part of the application for certification, contains an evaluation of an accident and resulting consequences caused by an Evaluation Basis Earthquake (EBE). The EBE chosen for the analysis is an earthquake with an expected return period of 250 years. This earthquake produces a peak ground acceleration of 0.05g at the site. The conclusions of this evaluation are discussed in Chapter 5, Accident Analysis.

In evaluating soil liquefaction potential at PORTS, three extensive exploration and laboratory testing programs (data sets) have been completed at the site, with the total number of exploratory borings approximately 960. These borings and accompanying laboratory test results were used at the site to analyze the response of soil to ground shaking caused by earthquakes.

The laboratory classification tests, shear strength tests, and consolidation test data were used to define the general engineering characteristics of the soil. Analysis of the data indicates that there is a low potential for soil liquefaction at the site, even in the unlikely event of the occurrence of an earthquake of magnitude 5.25 with a maximum peak ground acceleration of 0.15g. Consequently, settlement in the site area due to liquefaction is unlikely.

Chapter 3 ORGANIZATION AND ADMINISTRATION

The regulations in 10 CFR §76.35(a)(7) require that the SAR contain "A description of the management controls and oversight program to ensure that activities directly relevant to nuclear safety and safeguards and security are conducted in an appropriately controlled manner that ensures protection of employee and public health and safety and protection of the national security interests." Chapter 6 of the SAR describes the organization and management controls utilized by USEC to meet this requirement.

The characteristics of an organization which can safely handle special nuclear and source material include a clear assignment of responsibilities, including: (1) responsibility for the various components of safety; (2) an effective safety review system; (3) a training program for proper operation and proper conduct of the safety components; (4) clear and accurate procedures; (5) a means of knowing compliance with the rules and procedures with a means of assuring necessary compliance; and (6) an investigation process to enable understanding and promote fixing of significant institutional problems. The following sections briefly describe the organization and management controls in place at USEC.

3.1 Organization

USEC is a wholly-owned Government corporation which was established by the Atomic Energy Act, as amended. Members of the USEC Board of Directors were appointed by the President and confirmed by the U.S. Senate. Corporate offices are currently located in Bethesda, Maryland. USEC has hired a contractor, Lockheed Martin Utility Services, Inc., (LMUS) to operate the plants. USEC retains responsibility for the safe operation of the facility. USEC approves the management structure and key positions, assignment of individuals to key positions, and qualifications, responsibilities and authorities for key positions.

3.1.1 Safety Responsibilities

USEC and LMUS have established an organization that has independent chains for the safety functions. The organization is discussed in §6.1 of the SAR. TSR 3.3 requires USEC to use qualified individuals in facility positions, and to meet the responsibility and qualification requirements described in the SAR for the key staff positions. TSR 3.2.1 requires USEC to establish and define the lines of authority, responsibility, and communication. The TSR also provides for the safety functions having organizational freedom to ensure independence from operations. Figure 3.1-1 shows the organization structure.

The Executive Vice President, Operations, has overall responsibility for safe operations of the facility. This position has the authority to direct the General Manager to place the facility in a safe condition. Any operation directed to be shut down by the Executive Vice President cannot be restarted without his/her concurrence. This position is appointed by the USEC Board of Directors.

The Vice President, Production, has overall responsibility for all activities within the USEC production organization, including the functions of operations, maintenance, plant support,

engineering, transportation, materials handling and storage, and industrial, radiological, and nuclear safety. The individual in this position has shutdown authority and must concur on the restart decision. The Vice President, Production, is appointed by the USEC Board of Directors.

The USEC Nuclear Regulatory Assurance and Policy Manager is responsible for the management of USEC nuclear regulatory assurance functions, which includes the quality assurance policy. This position is responsible for the USEC Quality Assurance Plan and for determining the status, adequacy, and effectiveness of the Quality Assurance Plan. This position is independent from production and reports to the Executive Vice President, Operations.

The USEC Safety and Health Assurance and Policy Manager is responsible for plant fire and police services, nuclear material control and accountability, security, and emergency management activities, and for the development and management of the related corporate policies. This position is independent from production and reports to the Executive Vice President, Operations.

The USEC Environmental Assurance and Policy Manager is responsible for the development and management of corporate environmental and waste management policies and for these activities at the plants. This position is independent from production and reports to the Executive Vice President, Operations.

The positions discussed above are all located at the headquarters office; the positions discussed below are located at the PORTS site.

The Safety, Safeguards and Quality Manager is a USEC employee but is located at the site. He/she reports to the Executive Vice President, Operations. This position has the responsibility for the oversight of plant operations to ensure that the health and safety of the public and workers are adequately protected, to ensure compliance with safety, safeguards, and quality requirements, and to ensure implementation of USEC policies, procedures, and management expectations. This position manages the Safety, Safeguards and Quality Office and directs plant quality assurance functions involving audits and oversight of plant operations as well as a nuclear safety assurance function. He/she is also responsible for nuclear material control and accountability. This position has shutdown authority

The Safety, Safeguards and Quality Manager has an extensive role in the organization. This position serves as the on-site presence for USEC and will be responsible for monitoring activities for USEC. This position is also responsible for the quality assurance program and the nuclear material control and accountability program. The Quality Assurance Program (QAP) is in the early stages of development and will require a great deal of involvement. USEC needs to ensure that the support staffing is adequate to implement the necessary programs. This is an area in which the NRC staff has interest and will be carefully monitored through the observation and inspection program.

The General Manager is responsible for the safe operation of the plant, for compliance with all applicable NRC regulatory requirements, and for adherence to applicable USEC policies.

He/she is responsible for production, training and procedures, site and facilities support, engineering, transportation, materials handling and storage, occupational, environmental, and nuclear safety. The General Manager also has responsibility for the primary day-to-day interface with the NRC for issues of adequate safety/safeguards and regulatory compliance. The General Manager has shut down and stop work authority for all portions of the plant. This position also has startup authority, concurrence must be obtained for those operations shut down by the Executive Vice President, Operations; Vice President, Production; or the Safety, Safeguards and Quality Manager. The General Manager is a LMUS employee and reports to the Vice President, Production. The position is appointed by the President of LMUS with concurrence by the Vice President, Production, and the Executive Vice President, Operations.

The Enrichment Plant Manager is responsible for the day-to-day production activities at the site including operations, maintenance, work control, and production support. This position has the same shutdown/restart authority as the General Manager, except that the concurrence of the General Manager must also be obtained. The position is appointed by the General Manager with concurrence by the President, LMUS, and the Vice President, Production.

The Operations Manager is responsible for the safe operation of the enrichment cascade, plant utilities, chemical services, and feed and withdrawal facilities. This position may act for the General Plant Manager and the Enrichment Plant Manager. The Operations Manager has shutdown authority for that part of the operations for which he/she has responsibility.

The Maintenance Manager is responsible for providing safe and reliable performance of preventive, predictive, and corrective maintenance on production facilities and equipment. This position reports to the Enrichment Plant Manager.

The Production Support Manager is responsible for the technical functions in direct support of production. This includes the health physics program, laboratory operations, quality control, and the waste management program. He/she has the authority to stop work or shutdown operations in areas for which he/she has responsibility. This position reports to the Enrichment Plant Manager.

The Environmental Safety and Health Manager is responsible for establishing and implementing the environmental monitoring program, environmental protection programs, and the industrial and chemical safety programs. This includes activities associated with environmental compliance, occupational safety and health, industrial safety, chemical safety, and industrial hygiene. This position reports to the General Manager and has stop work and shut down authority for activities that could cause environmental, safety or health concerns.

The Engineering Manager is responsible for engineering activities in support of operations, including design, fabrication, and construction of plant modifications or additions; the configuration management program; systems and reliability engineering; and nuclear safety. This individual reports to the General Manager.

The Site and Facilities Support Manager reports to the General Manager and is responsible for plant fire and police services, security, non-production related facility maintenance, and the shared site program.

The Training and Procedures Manager is responsible for preparation, presentation, and recording of employee orientations and for technical and qualification training programs development and implementation. He/she is also responsible for the development and implementation of the procedures management program. This position reports to the General Manager.

The Plant Shift Superintendent Manager coordinates the activities of the plant shift superintendents and reports to the Enrichment Plant Manager. He/she provides technical and administrative support for the plant shift superintendents. He/she also has responsibility for emergency management.

The Plant Shift Superintendent represents the General Manager, and has the authority and responsibility to make decisions as necessary to ensure safe operations, including stopping work and placing the plant in a safe condition. He/she can also authorize restart after shutdown for non-routine reasons; however, approval of the Enrichment Plant Manager is necessary for operations shutdown by upper management. This position is responsible for accumulation and dissemination of information regarding plant activities. The Plant Shift Superintendent serves as incident commander during plant emergencies and is responsible for making event notifications. He/she has the authority to act for the General Manager and Enrichment Plant Manager, in their absence, regarding operational matters.

TSR 3.1.1 assigns corporate responsibility for overall GDP safety to the Executive Vice President, Operations. TSR 3.1.2 assigns responsibility for the overall safe plant operations to the General Manager. The Plant Shift Superintendent responsibilities are assigned by TSR 3.1.3. The Plant Shift Superintendent is responsible for the operational aspects of the plant and for the central control room command function. TSR 3.1.4 assigns the Division Managers responsibility for operations conducted within their facilities. These TSRs assigning responsibility are similar to the Westinghouse Standard Technical Specifications. The staff concludes that the organizational structure and assignment of responsibilities are consistent with good industry practice, meet the requirements of 10 CFR Part 76, and are, therefore, acceptable.

3.1.2 Technical Qualifications

The regulations in 10 CFR §76.35(a)(3) states that the SAR must include: "The qualifications requirements, including training and experience, of the Corporation's management organization and key individuals responsible for safety in accordance with the regulations in this chapter." Section 6.1.1 of the SAR describes the minimum qualifications needed for the key positions. It is the responsibility of USEC to ensure that individuals in these positions meet the qualification requirements. TSR 3.3 requires facility positions to be filled by individuals whose experience/training qualify them for the position. The minimum qualifications for key positions are described in the following paragraphs.

Executive, Vice President, Operations
The Executive Vice President, Operations, shall have a bachelors degree or equivalent technical experience, 10 years of management experience, and 6 years of nuclear experience.

Vice President, Production
The Vice President, Production, shall have a bachelors degree in engineering or the physical sciences or have equivalent technical experience. He/she shall also have 6 years of technical nuclear experience and 6 years management experience (may be concurrent).

Nuclear Regulatory Assurance and Policy Manager
The Nuclear Regulatory Assurance and Policy Manager shall have a bachelors degree or equivalent technical experience and 4 years of nuclear experience.

Safety and Health Assurance and Policy Manager
The Safety and Health Assurance and Policy Manager shall have a bachelors degree or equivalent technical experience and 4 years of nuclear experience.

Environmental Assurance and Policy Manager
The Environmental Assurance and Policy Manager shall have a bachelors degree or equivalent experience and 4 years of environmental management experience.

Safety, Safeguards and Quality Manager
The Safety, Safeguards and Quality Manager shall have a technical degree. He/she shall also have 15 years of nuclear experience with 3 years of management experience in quality assurance, nuclear safety oversight, engineering and technical support, or regulatory affairs. Either this position or a quality assurance manager that reports to this position must have a minimum of one year experience in quality assurance.

General Manager
The General Manager shall have a bachelors degree in engineering or the physical sciences or equivalent technical experience. He/she shall have 6 years of nuclear experience and 6 years of management experience (may be concurrent).

Enrichment Plant Manager
The Enrichment Plant Manager shall have a bachelors degree in engineering or the physical sciences or equivalent technical experience. He/she shall have 6 years of nuclear experience and 6 years of management experience.

Operations Manager
The Operations Manager shall have a bachelors degree in engineering or the physical sciences or equivalent technical experience. He/she shall have 4 years of nuclear experience with at least 6 months in a gaseous diffusion plant.

Maintenance Manager
The Maintenance Manager shall have a bachelors degree in engineering or the physical sciences or equivalent technical experience. He/she shall have 4 years of nuclear experience with 6 months in a gaseous diffusion plant.

Production Support Manager
The Production Support Manager shall have a bachelors degree in engineering or the physical sciences or equivalent technical experience. He/she shall have 4 years of nuclear experience and 6 months in a gaseous diffusion plant.

Environmental Safety and Health Manager
The Environmental Safety and Health Manager shall have a bachelors degree in engineering or safety disciplines, the physical sciences or environmental sciences, or equivalent technical experience. He/she shall have 4 years of nuclear experience with at least 6 months at a gaseous diffusion plant.

Engineering Manager
The Engineering Manager shall have a bachelors degree in engineering or the physical sciences and 4 years of nuclear experience. He/she shall also have at least 6 months experience in a gaseous diffusion plant.

Site and Facilities Support Manager
The Site and Facilities Support Manager shall have a bachelors degree or equivalent technical experience and 4 years of nuclear experience with at least 6 months at a gaseous diffusion plant.

Training and Procedures Manager
The Training and Procedures Manager shall have a bachelors degree or equivalent technical experience. He/she shall also have 4 years of experience in nuclear facilities.

Plant Shift Superintendent Manager
The Plant Shift Superintendent Manager shall have a bachelors degree or equivalent technical experience and 4 years nuclear experience. He/she shall have at least 6 months experience at a gaseous diffusion plant.

Plant Shift Superintendent
The Plant Shift Superintendent shall have a bachelors degree in engineering or the physical sciences or equivalent technical experience and 4 years experience at a gaseous diffusion plant, or a high school diploma plus 12 years experience at a gaseous diffusion plant.

Operations and Maintenance Supervisors
First-line operations supervisors shall have a high school diploma and 3 years of plant operations experience with at least 6 months in a gaseous diffusion plant. First-line maintenance supervisors shall have a high school diploma and 3 years of plant maintenance experience with at least 3 months in a gaseous diffusion plant.

The minimum technical qualifications provided in the application are sufficiently detailed to enable the staff to determine that the established technical qualifications are consistent with good industry practice, meet the requirements of 10 CFR Part 76, and are, therefore, acceptable.

3.2 Safety Review Systems

The regulations in 10 CFR §76.68(a) require that plant changes must be approved by a safety review committee. USEC has established a safety committee to assist in the oversight function required by 10 CFR §76.35(a)(7) and to meet the requirement in 76.68(a). The safety committee, which is the Plant Operations Review Committee (PORC), is described in SAR Section 6.2 and in TSR 3.10. The PORC functions in an advisory role and supports the General Manager. The PORC is the main committee established by USEC. USEC has also established an as-low-reasonably-achievable (ALARA) subcommittee and may from time to time establish other subcommittees to provide assistance in conducting the reviews and assessments required by the PORC. The following paragraphs describe the PORC and the subcommittees.

3.2.1 Plant Operations Review Committee

The PORC activities are described in TSR 3.10. The TSR establishes the membership, qualifications, meeting frequency and quorum, functions, responsibilities, and required records. The PORC performs multi-discipline reviews of plant activities to ensure that the day-to-day and proposed activities are conducted in a safe manner. The General Manager approves the procedure implementing the PORC activities. The Safety, Safeguards, and Quality Manager is responsible for auditing and oversight of PORC activities.

The PORC responsibilities include reviews of: 1) all proposed procedures and intent changes as required by TSR 3.9.2; 2) all proposed changes to the SAR; 3) all proposed changes to the plans submitted with the application; 4) all proposed changes to the TSRs, the TSR basis statements, the Certificate of Compliance, or the Compliance Plan; 4) all proposed changes to the plant or the plant's operations that require a written safety analysis in accordance with 10 CFR §76.68; 5) all nuclear criticality safety evaluations and approvals; 6) all proposed requests for Enforcement Discretion, and 7) NRC-required event reports.

PORC membership is multi-disciplinary and shall include experience in the following functional areas: cascade and chemical operations, engineering, maintenance, nuclear safety, nuclear criticality safety engineering, radiological safety, quality assurance, safeguards, and chemical, industrial, and environmental safety. The members have a bachelors degree in engineering or the physical sciences or equivalent technical experience; and four years of nuclear experience with at least six months experience at a gaseous diffusion plant. The member representing nuclear criticality safety engineering must have the qualifications of a Senior Criticality Specialist. A quorum consists of the chair and six members. The quorum must include members with technical competence in operations, engineering, nuclear criticality safety engineering, radiological safety, and quality assurance. The PORC meets at least once per calendar month. Written records shall be maintained.

The current PORC charter has not been reviewed to ensure consistency with TSR 3.10 and some of the procedures are not in place. These items are addressed in Compliance Plan Issue 22, however, they will be completed prior to NRC assuming jurisdiction.

3.2.2 ALARA

The ALARA committee is a subcommittee of the PORC and is discussed in §5.3.1.2 of the SAR. The committee has no approval, stop, or start work authorities; its authority is limited to review and recommendations. The ALARA committee functions to: 1) communicate management's commitment to the ALARA program; 2) monitor the implementation of the ALARA program and serve as advisor to plant management for maintaining occupational dose and environmental dose in accordance with ALARA principles; and 3) review, for the purpose of occupational dose and environmental dose reduction, proposed designs, practice, selected suggestions, and selected project schedules. Membership is designated by the PORC chair and represents various functional disciplines of the plant. The Health Physics Manager currently serves as the chair. The committee meets at least semiannually. Minutes and reports of the ALARA Committee are maintained as part of the PORC records package.

3.2.3 Subcommittees

The PORC chair may establish subcommittees to provide assistance in conducting reviews and assessments, however, PORC retains the overall responsibility for the required reviews. The PORC chair approves the subcommittee procedures, membership, and member qualifications.

The commitments for a safety committee have been reviewed and the staff concludes that the commitments are consistent with good industry practice, meet the requirements of 10 CFR Part 76, and are, therefore, acceptable. TSR 3.10 requiring USEC to have a PORC is acceptable.

3.3 Operations

Operations is one of the topics required by 10 CFR §76.87(c) to be included in the TSRs. Operations is discussed in SAR §6.5. The discussion includes such topics as normal operation, off-normal operations, emergency situations, key operations and facilities, control of hazards, operator and management responsibilities, permits and tagging, logkeeping, etc. TSR 3.19 commits USEC to establish, implement, and maintain the operations program described in the SAR and lists eight elements (i.e., shift operations.; cascade operations organization and administration; chemical/utilities/power organization and administration; operator responsibility, authority and shift routines; operations procedures; operator aids and system labeling; permits and logging; management monitoring of operations; and control of equipment) that must be addressed by the program.

The PORTS work force is divided into two primary groups, a day shift working primarily Monday-Friday and four rotating shifts that provide continuous coverage of plant operations. The gaseous diffusion process operates continuously. The day shift provides the administrative support, activities such as design and fabrication where around the clock effort is not required, procedures development, classroom training, planning, and preventive maintenance. The majority of the plant staff is assigned to the day shift. The shift organization has the prime responsibility for continuous plant operation and the evolutions, exchange of information, and response to abnormal and accidental conditions to ensure safe operation of the plant. Typically a shift: (1) provides oversight and direction for all plant operations; (2) monitors systems and equipment for proper performance; (3) conducts routine back shift maintenance and emergency equipment repair; (3) prepares equipment for day shift repair/preventive maintenance functions; and (4) responds to off normal or emergency situations.

TSR 3.2.2 establishes the minimum staffing requirements for each building in which operations having safety significance are carried out. TSR 3.2.2 also establishes overtime guidelines for plant staff who perform safety functions. At the time of certification, PORTS is not able to comply with its proposed overtime guidelines that limit overtime to no more than 24 hours in any 48 hour period, or no more than 72 hours in any 7 day period exclusive of shift turnover time. Compliance Plan Issue 37 addresses the overtime situation. PORTS commits to provide sufficient staffing and a revised TSR to the NRC to meet the above overtime guidelines by December 31, 1996. The staff finds this plan of action and schedule to be acceptable.

Each shift organization is composed of a Plant Shift Superintendent (PSS) and an assistant PSS; a Cascade Coordinator (CC) who directs overall cascade activities; first-line managers for the cascade buildings, power operations, chemical and utility operations, health physics technicians, Security Shift Commander, Fire Department Shift Commander, operators, security patrol officers, and firefighters.

The PSS provides a direct chain of command from the Plant Manager to the shift operating staff and serves as the senior shift manager in directing activities and personnel. During offshift emergency situations, the PSS acts for the Plant Manager until relieved. The PSS's normal watch station is in the X-300 Plant Control Facility (PCF). The PCF is the hub of

the plant operational activity. The overall UF_6 enrichment process is monitored at this location. Critical plant operations can be performed remotely from the PCF, key alarm systems are monitored, and plant communications systems as well as off-site communications capabilities are located in the PCF.

Plant operations, shift routines, and operator responsibilities are governed by a series of conduct of operations procedures. These procedures cover such items as alarm deactivation and control, shift routines, operating area logs and records, etc. A key part of the shift routine is operator rounds. In facilities requiring continuous operations, rounds are made a minimum of once per shift. Typical rounds consist of verifying proper system operating parameters, component alignment, and surveying the facility and equipment for abnormal conditions. Management provides day-to-day guidance on plant operations via operational memos, daily instructions, an long term orders; these guidance documents do not replace procedures. The problem reporting program provides the mechanism for employees to report to management regarding operational activities, equipment, procedures, etc., that may have potential for negative impact on health, safety, or security.

TSR 3.23 address worker protection from UF_6 process hazards. PORTS commits to establish, implement, and maintain worker protection measures to minimize the risk and mitigate the consequences of releases of UF_6 reaction products with moist air, and other associated process chemicals.

The staff finds the PORTS application on its program for plant operations and the associated TSRs to be acceptable for the protection of workers and the public.

Compliance Plan Issue 25 addresses aspects of the operations program that are currently not in place. Areas of noncompliance are in procedures and training (covered by Issues 30 and 26). PORTS will continue to use procedures and training that is currently utilized under DOE, until the new procedures and the training program are in place. The staff finds this to be acceptable.

3.4 Training

The regulations in 10 CFR §76.35(a)(5) requires USEC to submit a "training program that meets the requirements of §76.95." According to 10 CFR §76.95 a training program must be "established, implemented, and maintained for individuals relied upon to operate, maintain, or modify the GDPs in a safe manner. The training program shall be based on a systems approach to training."

USEC describes its training program in §6.6 of the SAR. TSR 3.4 requires USEC to establish, implement, and maintain the program as described in the SAR. The training program at PORTS consists of a number of training elements, some of which utilize the systems approach to training and some that do not. The following sections briefly describe the training program in place at PO'.TS.

3.4.1 Organization and Administration

The training organization consists of a centralized staff which reports directly to the training manager. The staff consists of technical trainers, administrative personnel, and mid-level managers who are directly responsible for assisting with the training program in the following functional areas: general employee training; operations and maintenance technical training; radiological worker training; health physics technician training; environmental, safety, and health training; subcontractor training; training record; and training instructor/developer qualification. Training staff are also assigned to interface with functional line managers to coordinate training for functional areas such as cascade operations, general plant support, health and safety, and chemical, utility and power.

USEC has established general employment policies and procedures to cover selection of personnel. Minimum qualification standards for key positions are documented and maintained. For positions requiring systems approach to training (SAT), the training requirements are defined in Training Development and Administrative Guides.

The training staff is responsible for Training Instructor/Developer Qualification. Training is provided to designated subject matter experts and the technical trainers who develop and/or conduct qualification training. Training is provided in the use of SAT in conducting analysis, design, development, implementation, and evaluation of plant training programs. The program includes initial training and periodic re-evaluation of skills and knowledge in both material development and/or instructional competency. Instructor/developers are required to spend a minimum amount of time working in the area they instruct. Instructors are expected to stay up-to-date on procedure/policy changes, modifications to TSRs, and lessons learned.

Training consists of initial and continuing training. The initial training is intended to instill an understanding of the fundamentals, basic principles, systems, procedures, and emergency responses involved for a particular job. This training includes the applicable process safety training such as technical requirements training and nuclear criticality safety training. The TSR training is designed to provide sufficient understanding of the safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls necessary for safe operation of the plant. Continuing training is provided to maintain and improve job-related knowledge and skills. This would include lessons-learned, procedure and equipment modifications and refresher training in such topics as emergency response, criticality safety, health physics, TSRs, general plant rules, etc.

Employees are trained commensurate with their duties. All new employees, subcontractors, and visitors who require unescorted plant access receive general employee training consisting of radiological safety, nuclear criticality safety, hazard communication, emergency preparedness, and other general topics. This training is required biennially.

Training attendance records, examinations, employee qualification record, and program records are maintained to document each employee's training. The training program records contain the job analysis data and history of the module.

3.4.2 Systems Approach to Training

The SAT is utilized at PORTS for those personnel who operate, maintain, or modify Q items or structures, systems, or components (SSCs) identified in the nuclear criticality safety approvals required to meet the double contingency principle. When a task is identified to operate, maintain, or modify a specific component in a system or process relied upon for safety, the training is developed using the SAT. The SAT approach is discussed in the following sections.

3.4.2.1 Systematic Analysis

A needs/job analysis is conducted using job incumbent/supervisor written surveys or the table-top method with subject matter experts. This is used to identify tasks affecting worker or public safety, safeguards, or protection of the environment as identified in the application. A facility-specific task list is developed and analyzed for the following positions: cascade operators, chemical operators, uranium material handlers, electricians, instrument and electronic mechanics, maintenance mechanics, system engineers, cascade controllers, plant shift superintendents, health physics technicians, laboratory technicians, environmental technicians, nuclear criticality safety engineers and specialists, and on-site transportation of UF_6. Each task is rated on degree of difficulty, importance, and frequency and from this analysis tasks are selected for training. The tasks selected for training are matrixed to the associated procedures and training materials. These matrixes and associated training materials are reviewed every 3 years and updated as necessary by changes in procedures, facility systems/equipment, or job scope.

3.4.2.2 Learning Objectives

Learning objectives for a specific task are established and incorporated in the training module developed for the task. Learning objectives state the knowledge, skills, and abilities the trainee must demonstrate to successfully complete the training. The objectives are updated as necessary based on changes in procedures and equipment.

3.4.2.3 Training Design and Implementation

Classroom lesson plans, on-the-job training (OJT) guides or other instructional materials are developed. These materials provide the guidance and structure to ensure consistent delivery of information from trainer to trainer and class to class. The classroom lessons are used mainly to provide cognitive learning on fundamentals, theory, basic operating and maintenance principles, individual systems, system inter-relations, safety requirements, and overviews of the GDP processes. Instructional materials such as video, computer-based training, and self-study are also sometimes used.

OJT is a method of providing in-field training and evaluation. The training is conducted in the actual work environment and demonstrates actual performance. This method is only implemented if management determines that manpower and operational conditions will not be impacted by the training activities.

The lesson plans, OJT, and other materials receive technical reviews by designated subject matter experts, instructional reviews by training personnel, and final approval for use of line management. Training materials are approved by responsible line management and the training staff before issuance.

3.4.2.4 Evaluation of Trainee Mastery

Trainee progress is evaluated by technical trainers and line management through several different methods such as written examinations, oral examinations, and practical tests to ensure mastery of the job performance requirements and learning objectives. Training courses consist of blocks for which evaluation can be performed for each individual block or for several blocks together. Remediation is provided as appropriate.

3.4.2.5 Evaluation of Program

Training effectiveness is evaluated through student and supervisor feedback. Students evaluate the course and instructor at the end of the training; and a post evaluation is conducted 3-6 months after completion of the training. First-line supervisors provide feedback on the student's actual performance. The program is also subject to assessment by line and training management through the self assessment program and through the quality assurance (QA) audit program. The feedback obtained is utilized to refine or revise the training programs.

The PORTS SAT program addresses the necessary elements of a good program and meets the requirements of the regulations. The effectiveness of the program will depend on how well it is implemented. Effectiveness will be evaluated as part of the inspection program. The staff has reviewed the training program and concludes that it is consistent with industry practices and meets the intent of the regulations and is, therefore, acceptable.

Although the program as described in the SAR is acceptable, the systems approach to training is still in the development stages and not fully implemented. This program is covered by Compliance Plan Issue 26. The current schedule has the initial round of training to be completed by June 30, 1997. The second round will be completed by December 31, 1997. The second round is for those positions that were not identified in the early stages as needing a SAT based program and include power operators, systems engineers, cascade coordinators, and nuclear criticality safety engineers/specialists. The training program for augmented quality (AQ) related activities will be completed by June 30, 1998. In the interim, USEC will continue to use the training program in place under DOE. This will be acceptable until the full SAT based program is complete.

3.5 Procedures

Although a procedures program is not specifically required by the regulations, it is considered an essential part of the management controls and oversight program required by 10 CFR §76.35(a)(7) and by NQA-1. USEC is committed to the use of approved and controlled written procedures to conduct nuclear safety, safeguards, and security activities for the protection of the public, plant employees, and the environment. Procedures prescribe the essential actions or steps needed to safely and consistently perform safety

related activities. The procedure program is described in §6.11 of the SAR. TSR 3.9 addresses the procedure program. The program is briefly described in the following paragraphs.

USEC uses a four level procedure hierarchy. Level 1 consists of policy statements issued by USEC and apply to all GDP personnel. Level 2 is standard practice procedures that apply to both sites or to more than one division. Level 3 procedures are issued at the organizational level and apply to more than one group. Level 4 procedures are those issued and applied within a group or subfunction. Section 6.11.4.1 and Appendix A to §6.11 of the SAR describes the minimum activities that shall be covered by written procedures. Topics covered are administrative procedures; system procedures that address startup, operation, and shutdown; abnormal operation/alarm response; maintenance activities that address system repair, calibration, inspection, and testing; emergency response; and any task that is described in, or implements a commitment that is described in the SAR, TSRs, and plans submitted with the application.

In Appendix B to §6.11 of the SAR, USEC has listed the specific subsections of American Nuclear Society (ANS) publication ANS 3.2-1994 that will be utilized in the procedure program. Procedures are developed or modified through a formal process. Procedure development, control, and use is a process that consists of nine basic elements: identification, development, verification, review and comment resolution, approval, validation, issuance, change control, and periodic review. PORC review is required for procedures required by the TSR and for intent changes to those procedures. All procedures are periodically reviewed to ensure their continued accuracy and usefulness. Emergency, Operating, Alarm Response and procedures dealing with highly hazardous chemicals are reviewed on a 1-year cycle. All procedures designated "in-hand", which involve liquid UF_6 handling activities, off-normal procedures, and nuclear material control and accountability procedures are reviewed on a 3-year cycle; all other procedures are on a 5-year cycle.

The procedure TSR 3.9 requires that written procedures shall be prepared, reviewed, approved, implemented, and maintained. The TSR covers the review and approval of procedures and allows for temporary changes. The procedure program as described in §6.11 and the TSR have been reviewed by the staff, the program is consistent with good industry practice and is, therefore, acceptable.

USEC is currently in the process of a procedure upgrade program; not all of the procedures needed to implement programs in the application have been prepared or upgraded and approved. This issue is addressed in Compliance Plan Issue 30. The site will continue to use existing procedures until new or revised procedures are complete. Procedures that contain action statements and operating limits from the TSRs will be in place prior to NRC taking jurisdiction. Level 2, 3, and 4 procedures that are related to Q and AQ-NCS (nuclear criticality safety) items will be completed by December 31, 1996, for AQ-NCS and by March 31, 1997, for Q; those implementing AQ and NS (non-safety) items will be completed by December 31, 1997. This approach and schedule is acceptable to the staff.

3.6 Human Factors

Human factors is not addressed in the regulations. However, USEC has proposed a human factors program. PORTS incorporates human factors considerations in engineering design work associated with new equipment and facility modifications; preparation, validation, and use of procedures; and in development of training and qualifications of personnel who operate, maintain, or modify structures, systems, and components relied upon for safety. Human factors is considered in problem reporting and investigation. Human actions required by the TSRs to prevent or mitigate accidents are systematically evaluated for human factor considerations on a 3-year cycle, including accessibility, visibility, ergonomic capability, suitability of the environment for the required activity, and interferences. This program will result in human factors considerations for those actions important to safety. The staff concludes that the program is acceptable.

3.7 Audits and Assessments

An audit and assessment program is not specifically required by the regulations but is considered part of the management controls and oversight program required by 10 CFR §76.35(a)(7) and the quality assurance program required by 10 CFR §76.35(d). PORTS has established a system of audits and assessments that are designed to ensure that the health, safety and environmental programs are adequate and effectively implemented. The audit and assessment program is described in SAR Section 6.8. TSR 3.5 requires USEC to implement the program described in the SAR and Quality Assurance Program. The program is designed to ensure comprehensive program oversight every 3 years.

Audits are conducted by the Safety, Safeguards and Quality organization as part of the QA program. The audit program is conducted in accordance with procedures and checklists by qualified auditors. Audits are used to verify the effectiveness of health, safety, and environmental programs and their implementation. Audits are also used to determine the effectiveness of the assessment program. Audit results are documented and reported as specified in plant procedures. The audit program is conducted in accordance with §2.18 and Appendix A of the QA Program.

The assessment program at PORTS consists of program assessments, divisional self assessments, management self assessments, and problem reporting. Personnel from the area being assessed may perform the assessment if they do not have direct responsibility for the specific area being assessed. Observations are resolved by the responsible division management. The assessment program is used as a tool to determine the effectiveness of the programs, compliance, and to verify corrective actions. Problem reports are issued on items of noncompliance or nonconforming conditions. In addition to the assessments, all plant employees are responsible for writing problem reports on safety, operating, and noncompliance items. Problem reports are screened and corrective actions taken as appropriate.

The staff has reviewed the PORTS audit and assessment program and concludes that it is consistent with good industry practice and is, therefore, acceptable.

PORTS does not have all the elements of the assessment program in place; this issue is addressed in Compliance Plan Issue 27. The procedures for the organizational assessment that specify the criteria and guidance for the assessments will be implemented before NRC assumes jurisdiction. This is acceptable to the staff.

3.8 Quality Assurance

The regulations in 10 CFR §76.35(d) and §76.93 require USEC to submit a QA Program that satisfies "each of the applicable requirements of the American Society of Mechanical Engineers (ASME) NQA-1-1989" or "acceptable alternatives to the applicable requirements." The regulations require USEC to "execute the criteria in a graded approach to an extent that is commensurate with the importance to safety." USEC submitted the QAP with the application. The QAP establishes the minimum requirements for those items, activities, and services within the scope of the QAP. USEC has committed to meet the Basic Requirements and Supplementary Requirements of ASME NQA-1-1989 or has committed to alternatives acceptable to the NRC.

3.8.1 QA Organization

The Safety, Safeguards, and Quality Manager is responsible for ensuring implementation of the QAP activities at the site. This manager directs PORTS quality assurance functions involving audits and oversight of plant operations. The QA organization is divided into three groups: Independent Assessment, Quality Systems, and Nuclear Safety Assurance. Independent Assessment performs audits and assessments of activities that affect safety, quality, and the environment. Independent Assessment also ensures the effectiveness of corrective actions. Quality Systems maintains the QAP, prepares procedures, reviews other procedures for inclusion of quality requirements, and participates in operational readiness reviews. Nuclear Safety Assurance performs independent engineering assessments, provides nuclear safety oversight, and performs selected operational experience reviews to determine affect on plant safety. In addition, there are other positions in the organization, particularly the Engineering Manager, that have QA responsibilities. Personnel responsible for QA activities are acceptably independent from operations personnel.

3.8.2 Quality Assurance Program

USEC has established a QAP that has three categories: Q, AQ, and NS. SSCs are categorized as Q, AQ, or NS by Engineering. The requirements of the main body of the QAP apply to the Q items and activities. Appendix A of the QAP defines the extent to which the QAP applies to AQ items and activities. Appendix A, Section 1, describes the QAP for Nuclear Criticality Safety (AQ-NCS) items and activities required to meet the double contingency principle as discussed in SAR §3.15. Appendix A, Section 2, describes the QAP for other AQ items and activities. Appendix A, Section 3, describes the QAP for other specified AQ structures. The formal QAP is not applied to NS items. SAR §3.15 lists the systems and boundaries for the Q and AQ items, except for AQ-NCS items that are listed in Appendix A of the QAP.

The requirements of the QAP apply to activities affecting the ability of Q and AQ SSCs to perform their intended function. These activities include designing, purchasing, fabricating, handling, receiving, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, and modifying. The QAP is implemented through policies, procedures, instructions, specifications, drawings, procurement documents, contractual documents, and other documents. These documents provide measures that ensure that QA activities are planned and accomplished to meet the goals and objectives of the QAP.

Procedures that implement the requirements of the QAP are reviewed by the affected organizations and authorized by the responsible manager.

The QAP requires the establishment of an indoctrination and training system to provide confidence that proficiency is achieved and maintained by PORTS personnel in the performance of their quality affecting activities. The QAP acceptably describes the qualification of personnel who verify the quality of items and work activities at PORTS. Quality is verified through surveillance, inspection, testing, checking, and auditing of items and work activities. The QAP requires that quality of items and work activities be verified by qualified personnel who are not responsible for performing the actual work. Quality verification activities are performed in accordance with procedures, instructions, and/or checklists by personnel who have been qualified in accordance with applicable codes, standards, or company training programs. Training and qualification of other personnel is addressed in Section 3.5 of this Compliance Evaluation Report (CER).

The QAP establishes a comprehensive audit system to ensure that the QAP requirements and related supporting procedures are effectively and properly implemented during operations. Audits will include an objective evaluation of QA practices, procedures, and instructions; work areas, activities, processes, and items; the effectiveness of implementation of the QAP; and conformance with policy directives.

The QAP requires documentation of audit results and review by the management personnel who have responsibility in the area audited to determine and take appropriate corrective action as required. Re-audits or surveillances are performed to determine that deficiencies have been effectively corrected and that the corrective action precludes repetitive occurrences. Audit findings are provided to the General Manager, the USEC Nuclear Regulatory Assurance and Policy Manager, and the Executive Vice President, Operations.

On the basis of its review, the staff concludes that the QAP for PORTS is structured in accordance with 10 CFR §76.93 and ASME NQA-1-1989. The QAP forms the foundation for the overall QA and describes how the requirements of 10 CFR §76.93 and ASME NQA-1-1989 are satisfied. The QAP and its implementing procedures control quality-related activities involving Q and AQ items to satisfy the requirements of 10 CFR §76.93.

The staff concludes that the QAP is consistent with good industry practice, meets the requirements of 10 CFR §76.93 and the basic and supplemental requirements contained in ASME NQA-1-1989, and is, therefore, acceptable.

The QAP is not fully implemented at the site and the program is covered by Compliance Plan Issue 32. This is acceptable to the staff.

3.9 Event Reporting and Investigation

The regulations in 10 CFR §76.120 and other applicable sections referenced in 10 CFR §76.60 identify the reporting requirements for the GDPs. The PORTS event reporting and investigation program is described in SAR Section 6.9. In addition to the requirements for oral notifications and written reports, USEC is required to determine root causes, adequate corrective actions, and lessons learned.

USEC committed to an event reporting and investigation program which is part of its management controls and oversight program. The program includes identification and categorization of events, NRC notifications, and an analysis to determine root causes, corrective actions, and lessons learned.

The program appears adequate to ensure that abnormal conditions are identified, reported, and investigated. Plant personnel are required to report to their line manager or PSS abnormal events or conditions that may affect plant safety. The PSS is responsible for categorizing and reporting events to the NRC. Table 6.9-1 of the SAR lists NRC event reporting requirements that USEC is committed to. It includes reporting criteria and reporting timeframe. The table contains reporting criteria which are not in the regulations, but USEC has agreed to report such as criticality reporting in accordance with NRC Bulletin 91-01 and safety system actuation reporting. Reports of classified information are done in accordance with the Security Plan for the Protection of Classified Matter. PORTS plans to review system designs where operational trips and alarms coincide with the setpoints for safety system actuation based on the same monitored parameter and the same equipment actuated. The results of this review will be submitted for NRC review and approval on October 1, 1996. In the interim, the trips and alarms will be treated and reported as an actuation of a system. This is acceptable to the staff.

An investigation will be conducted for each reportable event in accordance with written procedures. The Manager, Nuclear Regulatory Affairs, will form the investigation team and ensure the selection of investigator(s) and their qualification. The investigation includes, at a minimum, analyzing available information, interviewing involved individuals, identifying root causes, and developing corrective actions. Documentation related to the event will be retained in accordance with record retention requirements described in SAR Section 6.10.

The responsible management will develop and approve corrective actions to address root causes. The PORC will review and concur with the corrective action plan. The schedule for implementing corrective action is defined in each corrective action plan. These actions are entered into a commitment management database to ensure proper implementation and closure. The management will verify the completion of the corrective action before its closure in the database.

Lessons learned information from observations, events, problem reports, and experiences from GDPs and other related industries are reviewed and communicated within the plant

and between the two GDPs. Management reviews the information and determines if further action needs to be taken.

The staff has reviewed the PORTS event reporting and investigation program and concludes that it is consistent with good industry practice and is acceptable. As addressed in the Compliance Plan Issue 28, the procedures to implement this program have been completed for training purposes and training on these procedures will be completed by the time NRC takes jurisdiction over the GDPs. The procedures will become effective on the day NRC assumes regulatory oversight of PORTS. This is acceptable to the staff. Until NRC assumes jurisdiction, PORTS will continue to follow established DOE criteria in event reporting and investigation.

3.10 Record Management

A records management program is considered part of the management controls and oversight program required by 10 CFR §76.35(a)(7) and a necessary part of the QAP. The PORTS records management and document control programs are described in SAR §6.10. USEC identifies record retention period and commits to these programs in TSR 3.24.

USEC committed to centralized records management and document control (RMDC) programs. The records must be legible, readily retrievable, protected from damages, and retained for a certain timeframe. The RMDC programs are implemented through procedures. These programs address the criteria for the handling, verification, identification, authentication, indexing and filing, retention and disposition, correction, protection, storage, transmittal, retrievability, distribution, and assessment of records and controlled documents. The Manager of Documents and Records is responsible for the implementation of the RMDC programs.

The RMDC programs appear to be adequate. The programs ensure that records and documents important to safety and safeguards and security will be controlled, maintained and distributed according to written procedures. The programs also consist of instructions for special conditions, such as handling contaminated records and classified matters.

The Records Management program contains requirements for access control to ensure that only authorized personnel have the access to records. The record storage areas are fire rated and can protect and preserve records from loss, theft, tampering, unauthorized access, damage, and deterioration.

The use of computer codes and computerized data in the RMDC programs is controlled and maintained according to written manual and procedures. It addresses both access security and virus prevention. Precautions are taken to ensure validity of computer codes and data.

The QAP ensures that USEC's suppliers establish a comparable records management program. It also ensures that changes and corrections to records and documents are reviewed and approved by authorized personnel and are distributed and used at the locations where the prescribed activity is performed.

The staff has reviewed the RMDC programs and has determined that the programs are consistent with good industry practice and are acceptable. As described in Compliance Plan Issue 29, USEC has not completed the development and implementation of administrative and technical procedures for the RMDC programs. The RMDC procedures will be updated along with other procedures in the nuclear safety procedures upgrade project. Pre-existing records and documents will be turned over and incorporated into a centralized records management system via a turnover schedule by December 31, 1998. This schedule is acceptable to the staff. The records and documents needed to demonstrate that the facility can be operated and controlled within the operational envelopes specified by the application for NRC certification are either retrievable or are being recreated. Legacy records and documents, which are not readily retrievable at the time of certification, will be retrieved, updated, or re-created when they are needed for the evaluation of proposed plant changes pursuant to 10 CFR §76.68.

3.11 Maintenance

3.11.1 General

The description of the PORTS Maintenance Program is contained in the PORTS SAR, Section 6.4. The evaluation of the Maintenance Program is based on a review of SAR and discussions with USEC representatives. The Maintenance Program has been assessed to determine if it complies with the requirements of 10 CFR §76.35 "Contents of Initial Application."

3.11.2 Organization

The USEC Executive Vice President, Operations, has overall responsibility for safe operations of PORTS. The Vice President, Production, reports to the Executive Vice President, Operations, and has overall responsibility for production activities including operations, maintenance, plant support, and engineering.

The General Manager reports to the Vice President, Production, and is responsible for the safe operation of the plant and compliance with all NRC regulatory requirements. The Enrichment Plant Manager reports to the General Manager and is responsible for the daily production activities at the site including operations, maintenance, work control, and production support. The Maintenance Manager reports to the Enrichment Plant Manager and is responsible for providing preventive and corrective maintenance on facilities and equipment.

The Maintenance Unit is divided into five groups. These are Mechanical, Electrical, Instrument and Controls, General Production Service Shops, and Calibration and Electronics. Each group manager reports to the Maintenance Manager and is responsible for the performance of maintenance within the group's scope.

3.11.3 Corrective Maintenance

Corrective maintenance comprises actions intended to check, troubleshoot, and repair equipment that has degraded or failed. Corrective maintenance is initiated by issuance of a

work order that identifies and prioritizes the need for maintenance. A job package is assembled that identifies necessary materials and schedules, identifies required support services, and provides for post maintenance testing as required. Work is performed after a pre-job briefing and a completed maintenance is documented.

3.11.4. Preventive Maintenance

Preventive maintenance are tasks performed on a periodic basis to prevent failures, facilitate performance, and maintain life of equipment. Maintenance procedures identify and specify the schedule for preventive maintenance. The bases for tasks and their periods of performance are developed through a review of manufacturer recommendations, available industry standards, and historical operating information. Engineering (the design authority) approves and controls the documentation of preventive maintenance bases.

Preventive maintenance is scheduled according to plant conditions and the specified task frequencies. TSRs specify actions to be taken if scheduled maintenance activities are not completed within the required period for equipment relied upon for safety. For items not governed by TSR requirements, the Operations Manager and system engineer evaluate the effect of extending the scheduled period. This includes a determination of any compensatory actions and a rescheduling of the preventative maintenance. Permanent changes to preventive maintenance tasks and schedules require design authority approval.

3.11.5 Control of Work

Work control procedures apply to both corrective and preventive maintenance. Corrective maintenance is initiated by any manager identifying the need for maintenance by issuing a work order. The Work Control Supervisor reviews the work order to determine the need for a work package. Procedures in the work control program identify equipment that requires a work package before maintenance can be performed.

Work control procedures define the content of a work package. The minimum content for a safety system work package consists of the work order, a planning checklist, written work instructions, and a safety system data sheet. The package may also contain equipment specific procedures, quality control inspection requirements, nuclear criticality safety analysis, and health physics requirements. The package is developed using a planning sheet that lists the criteria for items and systems requiring System Engineering, Industrial Safety, Security and Fire Services, Health Physics, Nuclear Safety, and Waste Management approval. Engineering review and approval is always required for modification to plant equipment. The appropriate Maintenance Group Manger reviews and approves work packages prior to starting work. Authorization for release of equipment is required from designated operations managers before removing equipment from service.

3.11.6 Compliance Plan Items

The PORTS Compliance Plan, Issue 24, describes the aspects of the Maintenance Program that are not implemented as described in the PORTS SAR, Section 6.4. Areas not in compliance are:

1. Maintenance history and trend analysis program has not been developed.

2. All Q and AQ list items have not been identified.

3. Work package requirements for Q and AQ items have not been integrated into the Maintenance Program.

4. Development of work control procedures has not been completed.

5. Training for the above undeveloped procedures has not been completed.

The Compliance Plan states that continued operation is justified by several interim commitments. Work control procedures developed to meet the requirements of the DOE Regulatory Oversight Agreement and currently in place will continue to be followed. A minimum work package composition has been described. System engineers have been assigned responsibility for specific safety systems. The system engineers will provide technical support for maintenance activities including work package development, determination of post-maintenance testing requirements, observance of surveillance testing, and review of procedures.

3.11.7 Conclusion

The staff review of the PORTS Maintenance Program has verified that the requirements of 10 CFR §76.35 have been addressed. The staff concludes that the maintenance management organization provides the authority to carry out the functions of the Maintenance Management Program. The policies, procedures, and controls described in the SAR comply with the requirements of 10 CFR §76.35 with the exceptions noted in the Compliance Plan and 3.11.6 above. The staff finds the justifications for continued operation and the schedules for achieving compliance to be acceptable.

3.12 Configuration Management

3.12.1 General

The description of the PORTS Configuration Management Program is contained in the PORTS SAR, Section 6.3. The evaluation of the Configuration Management Program is based on a review of SAR and discussions with USEC representatives. The Configuration Management Program has been assessed to determine if it complies with the requirements of 10 CFR §76.35 "Contents of Initial Application," 10 CFR §76.36 "Annual Renewals," and 10 CFR §76.68 "Plant Changes." Guidance for the review was obtained from the draft 10 CFR Part 70 Standard Review Plan.

3.12.2 Organization

The USEC Executive Vice President, Operations, has overall responsibility for safe operations of PORTS. The Nuclear Regulatory Assurance and Policy Manager reports to the Executive Vice President, Operations. The Nuclear Regulatory Assurance and Policy Manager is responsible for nuclear regulatory assurance functions including configuration

management policy. The Vice President, Production, reports to the Executive Vice President, Operations, and has overall responsibility for production activities including operations, maintenance, plant support, and engineering.

The General Manager reports to the Vice President, Production, and is responsible for the safe operation of the plant and compliance with all NRC regulatory requirements. The Nuclear Regulatory Affairs Manager reports to the General Manager and is responsible for the management of commitments to NRC. The Nuclear Regulatory Affairs Manager is governed by the policies established by the Nuclear Regulatory Assurance and Policy Manager and reports to him on the status of compliance. The Nuclear Regulatory Affairs Manager is responsible for the maintenance of the TSR Program Plan documentation and the SAR Program Plan documentation, and for the procedures and training flowdown. The Engineering Manager reports to the General Manager and is responsible for design, fabrication, and construction of plant modifications and the Configuration Management Program.

3.12.3 Control of Items Relating to TSR Requirements

The Configuration Management Program requires that proposed changes to the plant, plant operations, training, and program plans are reviewed for their relationship to TSR requirements. Any modification involving a change to an item required by TSR requires NRC approval prior to implementing the change. Any such changes must also be approved by the PORC. Nuclear Regulatory Affairs controls the documentation and coordinates the implementation of changes to items required by TSR.

3.12.4 Control of Items Described in the SAR

The Configuration Management Program requires that proposed changes to the plant, plant operations, training, and program plans are reviewed for their relationship to descriptions provided in the SAR. The program requires that the changes meet the provisions of 10 CFR §76.68(a) if the changes are to be implemented without prior NRC approval. Revisions to the SAR that reflect any such changes will be provided to the NRC annually in accordance with 10 CFR §76.36. Nuclear Regulatory Affairs controls the documentation and coordinates the implementation of changes to items described in the SAR.

3.12.5 Physical Plant Change Control

Engineering (the plant Design Authority) controls changes to the physical plant and maintains the plant configuration as described in the SAR and as required by the TSRs. The Configuration Management Program has been documented in a site procedure that includes:

1. Identification of the structures, systems, equipment, components, and design features relied upon for safety and safeguards.

2. Assignment of organizational duties and responsibilities.

3. Administrative controls, procedures, and policies to maintain the plant baseline configuration.

Features controlled by the Configuration Management Program include those designated as Q or AQ items by the Quality Assurance Program.

Engineering has established the Change Control Board which provides a technical review of all proposed changes. The review determines the necessity of the change and the requirement for review and approval by the PORC. If the change is not a substitution, the proposal proceeds through the design modification process. An engineering group is assigned to process the change request. A design/project team is assigned that may include personnel from Engineering, Safety Analysis, Safety, Nuclear Criticality Safety, Operations, and Maintenance as appropriate. A safety evaluation is performed by Safety Analysis. Modifications are evaluated for any required changes to procedures, training, testing, or regulatory documentation.

3.12.6 Compliance Plan Items

The PORTS Compliance Plan, Issue 23, describes the aspects of the Configuration Management Program that are not implemented as described in the PORTS SAR, Section 6.3. Areas not in compliance are:

1. Flowdown of TSRs, SAR, and program plans into procedures and training requirements has not been completed.

2. Q and AQ lists have not been fully developed to identify items that fall within the scope of the program.

3. Design requirements for cranes and associated liquid-UF_6-handling equipment have not been identified.

4. Design requirements for Q list items not currently identified as safety systems have not been completed.

5. Nuclear Criticality Safety Approvals have not been reviewed for identification of AQ-NCS list items.

6. AQ list items for which design requirements must be developed have not been identified.

7. Development of records management and document control programs has not been completed.

8. Physical plant change control procedures have not been fully developed and implemented as described.

9. The program to assess the functioning of the Configuration Management Program has not been developed.

10. Personnel have not been trained on the above items which have yet to be developed.

The Compliance Plan states that continued operation is justified by several interim commitments. All requests for changes, other than like substitution, will obtain engineering services approval and will be reviewed by the Change Control Board. Design requirements will be developed/recovered on a case-by-case basis to support any change request review. Listings of all changes to plant and equipment safety systems will be maintained for each calendar year and will be available for review. Change control procedures currently in place to meet requirements of the DOE Regulatory Oversight Agreement will be continued until the development of configuration management program procedures is complete.

3.12.7 Conclusion

The staff review of the PORTS Configuration Management Program has verified that the requirements of 10 CFR §76.35, .36, and .68 have been addressed. The staff concludes that the configuration management organization is sufficiently independent from production management and provides the authority to carry out the functions of the Configuration Management Program. The policies, procedures, and controls described in the SAR comply with the requirements of 10 CFR §§76.35, 76.36, and 76.68 with the exceptions noted in the Compliance Plan and in 3.12.6 above. The staff finds the justifications for continued operation and the schedules for achieving compliance to be acceptable.

3.13 Management Controls

The regulations in 10 CFR §76.35(a)(7) require the SAR to contain a "description of the management controls and oversight program to ensure that activities directly relevant to nuclear safety and safeguards and security are conducted in an appropriately controlled manner that ensures protection of employee and public health and safety and protection of national security interests.".. USEC and LMUS have established management systems with associated policies, administrative procedures, and management controls to ensure protection of the health and safety of workers and the public, protection of the environment, and for the common defense and security. Management systems and programs are described in Chapters 5 and 6 of the SAR, in the TSRs, and in the program plans.

Preceding sections of this Compliance Evaluation Report discussed many of the programs that come under the consideration of management controls. Primary among these are an organizational structure that has clear assignment of responsibilities and independent reporting chains for the safety functions, PORC, QA, configuration management, an audit and assessment program, and an investigation and reporting process. The PORC provides the necessary review for management to make informed decisions. The audit and assessment program provide assurance that programs are being implemented in accordance with regulations and procedures. A QAP is in place to promote safe, reliable,

and efficient plant operation. PORTS investigates incidents to determine root cause and lessons learned. Items from the lessons learned are integrated into the procedures and training programs as appropriate. PORTS has a commitment tracking and corrective action management system that prioritizes plant actions consistent with their safety and safeguards significance. Once in place, these items along with a Procedures and Training Program, a Maintenance Program, the Configuration Management Program, and other programs will provide the necessary tools for USEC to operate in a safe, reliable fashion. Some aspects, such as PORC, engineering evaluations, and QA are still in the developmental stages. These activities will be closely followed by the NRC staff to ensure that the programs are being effectively implemented.

Some of the management control aspects are not currently in place and are addressed in the Compliance Plan. Items from the QAP, TSRs, and other requirements have not been flowed down to policies and procedures. Additionally, organizational responsibilities and authorities have not been flowed down to position descriptions. An initial flowdown will be completed by the time the NRC takes jurisdiction of PORTS. The final flowdown will not be complete until the procedure upgrade program is complete on December 31, 1997. This approach is acceptable to staff; the key aspects will be in place before NRC assumes jurisdiction. Items related to the Q list will be in place by the end of 1996. In the interim, USEC will continue to utilize the tools currently in place, including procedures, training programs, tracking systems, etc. The justification for continued operation and the schedule for completion is acceptable to the staff.

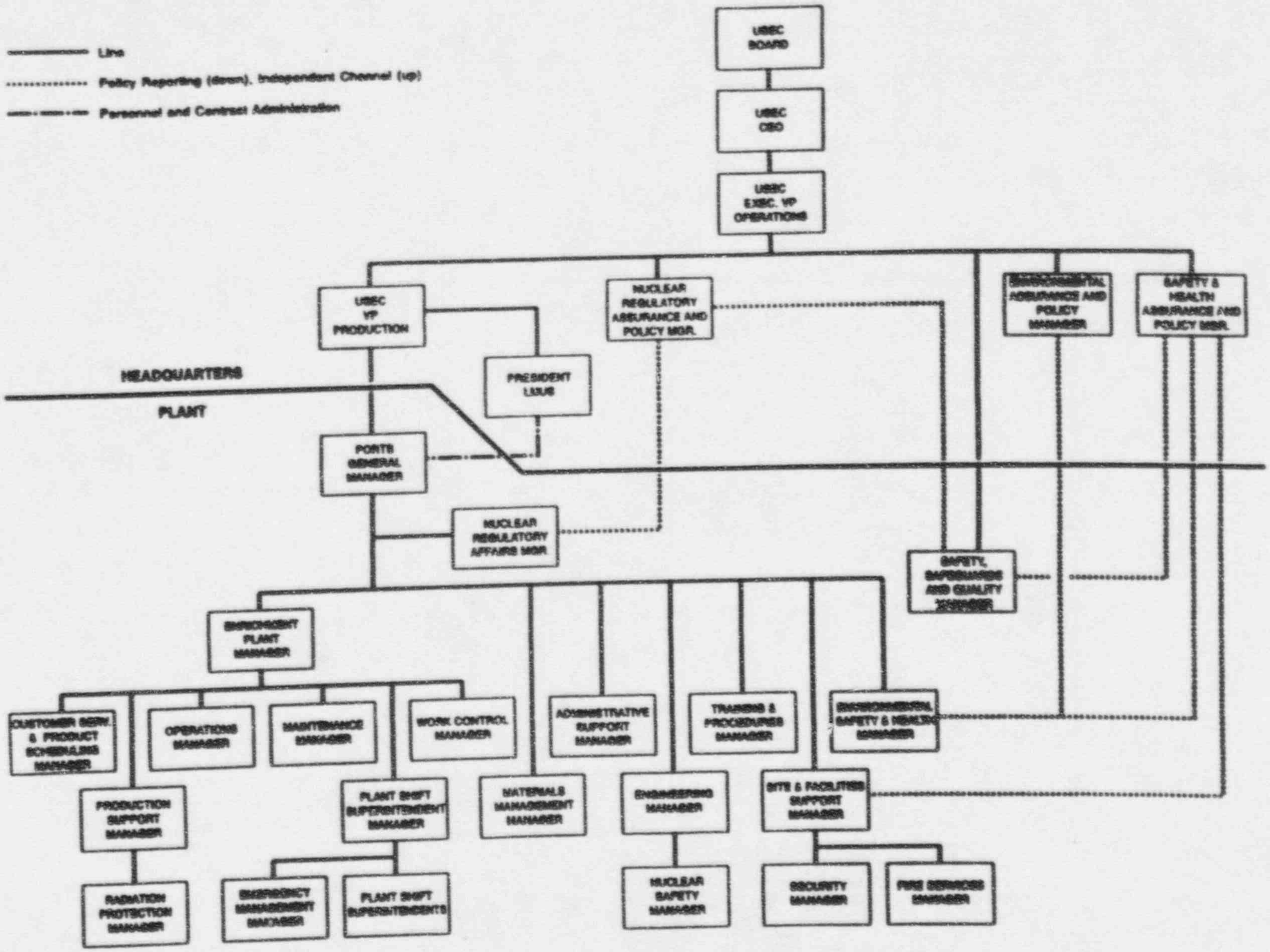

Figure 3.1-1 Uranium Enrichment Facilities Organization Chart

Chapter 4 FACILITY AND PROCESS DESCRIPTION

The regulations in 10 CFR §76.35(a)(8) require USEC to provide a "description of the principal structures, systems, and components of the plant." Chapter 3 of the SAR provides the facility and process description.

The PORTS is designed to separate a feed stream containing either naturally occurring proportions of uranium isotopes, or enriched product from the Paducah GDP, into a product stream enriched in the ^{235}U isotope and a tails stream depleted in the ^{235}U isotope. The chemical form of the working material of the plant, UF_6, does not require chemical transformations at any stage of the process. In the three primary steps of the process, UF_6 is volatilized from a feed cylinder, passed through the diffusion process, and condensed into product or tails cylinders. The description of the process presented in this chapter follows the flow of UF_6 from its receipt through processing to the disposition of product and tails material. The descriptions are drawn from the SAR. Additional detail and building and equipment drawings are provided in the SAR.

4.1 UF_6 Receipt and Feed

Feed material is received at PORTS in cylinders that contain 10 to 14 tons of UF_6 in the solid state. Cylinders containing UF_6 arrive at PORTS by rail or truck. Normal assay and Paducah product feed material is received at the X-343 Feed Vaporization and Sampling Facility. The feed cylinders are unloaded, inspected, weighed, and cold pressure checked. Some of the cylinders are liquified in an autoclave for sampling to ensure conformance with feed material specifications and for uranium accountability.

For sampling, the UF_6 cylinder is positioned in one of the 84-inch diameter autoclaves in X-343 or X-342A with the cylinder valve at the 12 o'clock position. The cylinder is connected to a manifold by means of a copper tubing pigtail. After pressure testing all connections, the UF_6 cylinder valve is opened, the emergency cylinder valve closer is attached, the autoclave shell is closed, and a hydraulically operated rotating ring seals the movable shell of the autoclave to the fixed head.

A vent admits steam into the autoclave and bleeds atmospheric air from the shell. Autoclave temperature is controlled by utilizing a cascade control scheme within a feedback loop controller. The steam pressure is limited to a maximum of 8 psig which corresponds to a saturated steam temperature of 235°F. A pressure transmitter monitors the cylinder pressure and closes the steam supply valves if the cylinder pressure exceeds 90 psia.

When the heating cycle is completed, the cylinder valve is closed, and steam is evacuated from the autoclave. The pigtail is purged and evacuated, the autoclave is opened, the pigtail is disconnected and the cylinder is rotated to the 3 or 9 o'clock position to permit liquid sampling. The pigtail is reconnected and pressure tested and the cylinder valve is partially opened before closing the autoclave shell. The steam block valve may be opened to maintain temperature during sampling.

During sampling, the inlet valve to the pipette in the sample cabinet is opened and the pipette is allowed to fill. The pipette sample evacuation valve is opened and the entire sampling system is flushed with liquid UF_6 to the evacuation drums. The evacuation valve is closed and the pipette is valved to an evacuation source to ensure all UF_6 is removed from the pipette and sampling manifold. All system valves are closed and the pipette inlet valve is opened to fill the pipette with liquid UF6. After closing the inlet valve, the sample is transferred to a sample container and the operation may be repeated to obtain duplicate samples. Following sampling, the pigtail is purged and removed from the UF6 cylinder using the procedure discussed previously. The UF_6 cylinder is then rotated to return the cylinder valve to the 12 o'clock position. After sampling, the cylinders are moved by crane to a storage yard where the cylinders are allowed to resolidify.

There are four feed autoclaves, with a 72-inch diameter, in the X-343 feed facility. Each is a cylindrical pressure vessel approximately 23 ft long which is mounted with its long axis horizontal. The autoclave shell moves approximately 15 ft from the fully closed to the fully opened position. The moveable shell of each autoclave is secured to the fixed head with a hydraulically operated flanged closure, which is sealed with a gasket.

The autoclaves in the feed vaporization facilities are designed to provide steam heating of 2.5-, 10-, and 14-ton cylinders of UF_6. The autoclaves are containment-type autoclaves with a maximum allowable working pressure (MAWP) of 163-165 psig. Autoclaves are designed to minimize standing water inventory to contain the products of a worst case UF_6 release. The feed vaporization autoclaves have a pressure relief valve that will relieve any pressure within the autoclave in excess of the MAWP of the autoclave to atmosphere through a vent line, which extends above the roof.

Each pigtail has an integral air operated isolation valve. When a cylinder is placed in the autoclave, the pigtail is connected and the air line attached to the isolation valve motor. The valve can then be closed from outside the autoclave by pressing the air motor start button. The air motor is not reversible, so the valve must be opened manually by an operator.

Service lines penetrate the autoclave head and are closed by redundant isolation valves that close automatically to prevent the escape of reaction products in the event of a UF_6 release inside the autoclave. The service lines that penetrate the head include the steam supply line, conductivity cell steam sample lines, air line, UF_6 feed line, condensate drain line, and thermal vent. The autoclave relief line connects to the rupture disk and relief valve. A pressure tap, and temperature probes also penetrate the stationary head.

Each autoclave is designed to withstand the reaction pressure developed by the exothermic reaction of UF_6 with the steam vapor and the condensate present. The maximum pressure generated by the reaction can be controlled by limiting the amount of water present. To ensure that the maximum allowable water inventory of the autoclave is never exceeded, redundant condensate level probes are mounted at the same elevation in the condensate drain slightly below the autoclave floor. Any accumulation of condensate in the drain line reaching the level of these probes automatically shuts the steam supply isolation valves and prevents the addition of more water.

In the event of a release, reaction products of UF_6 and water are detected by a conductivity cell and/or the pressure instrumentation, either of which effect autoclave containment by closure of isolation valves at the penetrations. The conductivity cell which constantly measures the conductivity of a sample of the autoclave atmosphere, is intended to detect small UF_6 releases that may not result in a pressure increase detectable by the high pressure isolation system. The cell is sensitive to the HF component of the UF_6 reaction with water. When triggered, this cell initiates a signal that closes all isolation valves.

For large releases the use of a pressure sensor provides a more rapid isolation than that obtained from a conductivity cell. For those releases having a large and rapidly rising pressure the autoclave isolation system is initiated by the pressure sensor.

As a protection against over pressure, each autoclave is provided with a rupture disc backed by a pressure relief valve. The relief system is set to relieve in accordance with ASME Code, Section VIII. Pressure sufficient to rupture the disc and open the relief valve will vent the autoclave to the atmosphere. This relief system would function only if there were an excessive amount of water in the autoclave at the time of a UF_6 release within the autoclave.

Autoclave temperature is maintained constant by an automated controller with a maximum of 235°F. A timed start-up system provides the warning mechanism to protect against a plugged valve or pigtail. If the cylinder pressure fails to reach 20 psia within an hour, the steam supply isolation valves are closed. Heating of feed cylinders is started approximately three hours prior to the estimated time that the cylinders will be needed for feed.

The UF_6 feed flow system maintains the desired flow rate to the cascade. Alarm conditions available at each feed station are: (1) indication that the feed control valve is fully open, (2) indication of low UF_6 header pressure, and (3) indication of high UF_6 header pressure.

Since material with different assay levels can be fed simultaneously from different feed stations, a feed header crossover system buffer is provided. The piping that connects feed lines carrying UF_6 of different assays is double blocked and maintained at a vacuum. The piping downstream that connects feed lines carrying UF_6 of different assays is also double blocked and is maintained at a positive pressure to prevent assay mixing.

Cylinder evacuation, or "heeling," is accomplished by connecting the cylinder to the evacuating header which provides a suction on the nearly empty cylinder. This will remove the residual UF_6 from the cylinder. Heeling can also be accomplished by connecting the nearly empty cylinder to other low pressure sources (e.g., cascade surge drums, purge and evacuation pumps, etc.). When the cylinder is acceptably heeled, the steam supply and the cylinder valve are closed. The autoclave pressure is reduced to atmospheric (indicated pressure < 0.5 psig) by jetting the autoclave through a steam venturi. After pigtail purging and evacuation, the empty cylinder is removed from the autoclave with the bridge crane and transferred to the storage yard.

The overhead bridge cranes handle only cylinders that are empty or contain solid UF_6. Each crane hoist has a shoe brake that is spring actuated in the event of a power loss. A geared up/down limit switch is connected to the cable drum to prevent exceeding limits for lowering or hoisting the load. When activated, it will stop the motor and activate the shoe brake. Each of these cranes uses an H-frame-type fixture to lift the cylinders with its single hook.

Autoclaves are connected to the steam supply, air supply, UF_6 jet exhaust system, and cascade feed station by piping which is enclosed in heated housings or steam-traced to prevent freeze-out of UF_6.

All autoclave piping to the outside containment valve is installed according to national standards and engineering piping specifications with a pressure rating of 200 psig minimum. Piping downstream of the containment valves is installed with pressure rating equivalent to its intended service.

UF_6 valves installed in the feed facilities are designed and tested to engineering specifications and have an externally buffered bellows stem seal backed up by a packing gland secondary stem seal. Pneumatically operated valves have air to open-spring to close-type actuators. The actuators on containment isolation valves are required to always fail to the safe position. Actuators are equipped with limit switches that indicate the open or closed position of the valve on a local panel. Valve bodies are of steel or monel construction, corresponding to pipe size material specifications.

Isolation valves are suitable for UF_6 gas or liquid service and a temperature range from 700°F to 3000°F. Pressure ratings are 200 psig for all valves that are used as primary containment valves. Valves located downstream from these valves are installed with a pressure rating sufficient for intended service.

The autoclave manual isolation system contains a manual push button at each autoclave and one button located in the X-343 area control room (ACR). When the button is pressed, each facility autoclave is placed into containment. The button is used upon confirmed UF_6 outleakage to mitigate the release.

The Q systems associated with the feed facilities are: UF_6 cylinder high pressure cutoff system, autoclave shell high steam pressure shutdown system, autoclave shell high pressure containment shutdown system, UF_6 cylinder high temperature cutoff system, high condensate level cutoff system, autoclave shell high pressure relief system, pigtail line isolation system, liquid UF_6 handling cranes, liquid UF_6 cylinder lifting fixtures, autoclave UF_6 cylinder pigtails, and the criticality accident alarm system.

4.2 UF_6 Enrichment

The PORTS enrichment facility consists of three cascade buildings. The cascade buildings are X-333 (640 "000" stages), X-330 (500 "00" stages and 600 "0" stages), and X-326 (720 "X-27" stages, 1440 "X-25" stages, and 180 purge cascade stages). Of the stages in X-326, 1680 have been retired in place. Thus PORTS currently has about 2400 active stages. The degree of isotopic separation is only about 0.2% per stage, therefore, about

1120 stages are used between the feed point and product withdrawal point to enrich uranium from normal feed to a product at 5 weight % ^{235}U. These stages are called the enriching section. About 620 stages strip the ^{235}U isotope from normal feed to a tails withdrawal assay of 0.2 to 0.3 weight % ^{235}U; these stages are called the stripping section.

A unit consists of 10 or 20 cells, depending on position within the cascade. There are eight "000" units in X-333 and five "00" and six "0" units in X-330. X-326 contains 3 "X-27" units, 6.5 "X-25" units, and a half unit purge cascade. Each stage contains a motor, compressor, converter, control valve, coolant system, and associated instrumentation. Each stage is connected to the next upper and lower stage throughout the cascade. The piping to a unit is arranged so that any unit or cell can be bypassed, without shutting down any other cells.

The stage converter is designed to support the barrier material in contact with the process gas stream, separate the enriched fraction of the process gas stream from the portion depleted in lighter isotope, and remove the heat of compression of the process gas in the converter by the use of cooling coils. The process gas enters the barrier region of the converter via a common inlet nozzle, where a portion of the process gas diffuses through the barrier tube wall and passes through the process gas cooler to the next compressor. The stream that diffuses through the barrier tube wall and is slightly enriched in the $_{235}U$ isotope is called the A-stream. The gas that does not diffuse through the barrier tube wall leaves the converter slightly depleted in the ^{235}U isotope (the B-stream). The A-stream is routed to the next higher stage compressor in the cascade, while the B-stream is routed to next lower stage compressor.

In order to efficiently operate the cascade, the stages near the feed point must have higher total flow rates than those near the withdrawal points. The flow taper required for efficient use of power is achieved by sequencing different equipment sizes and by varying the pressures across each size equipment. At the PORTS facility, 5 sizes of equipment are used in the cascade, and are designated, from largest to smallest, "000", "00", "0", X-27, and X-25.

Tie lines between buildings provide a means to move process gas from building to building without the need for withdrawal and feed facilities in each building. The tie lines are housed in elevated heated housings.

At those points in the cascade where there is a change in equipment size and a transition in pressure, the gas pressure can be increased by passing it through a booster station. These stations are also used at the product and tails withdrawal points and at the junction with the purge cascade. The compressors are either axial or centrifugal, depending on the flow rate and pressure ratio required. In general, axial compressors are used to boost the "A" (enriched) stream, while centrifugal compressors are used to boost the "B" (depleted) stream. An axial compressor is used to boost the bottom overlap to the upper cascade, while centrifugal compressors are used on the top overlap.

Surge drums and freezer/sublimer (F/S) units provide surge volume for fluctuations in inventory and power. Freezer/sublimer units are located in certain cells in the process

buildings and can be operated independently or in conjunction to remove excess UF_6 inventory, to conduct rapid power drops, to stop UF_6 outleakage, or to adjust the power load.

The F/S is cooled with R-114 creating a low pressure vessel into which UF_6 flows from the cascade B-line and solidifies on the heat exchanger tubes. A control valve on the inlet line is used to control the rate at which UF_6 is added to the vessel commensurate with the rate at which cascade power load is being decreased. The freeze rate is normally controlled between 60 and 100 lbs/min.

To remove UF_6 from the F/S and return it to the cascade, the freezing process is reversed. The vessel is heated with R-114 and the UF_6 changes phase from solid to gas (sublimation). F/S pressure is limited to less than the triple point by the cascade pressure to preclude UF_6 liquefaction during the sublimation process. The freezer/sublimer vessel and UF_6 piping are nickel-plated steel.

A number of functional trips have been incorporated into the freezer/sublimer system. A high UF_6 pressure alarm/trip is activated when the UF_6 pressure reaches 18 psia. A differential pressure trip will place the freezer/sublimer system in the hot standby mode should the R-114 decrease to within 2.5 psi of the recirculating cooling water (RCW) pressure, indicating a tube rupture within the R-114 condenser/reboiler. The high weight trip places the freezer/sublimer in the cold standby mode when the unit reaches approximately 9,000 lb of UF_6.

Each cascade building also has surge drums, which are used for evacuation, purging and cell treatment activities. The surge drums are large cylindrical vessels made of 5A-28SC steel mounted with their long axis horizontally. They are normally operated so that their pressure remains below atmospheric pressure. Each bank of drums is enclosed within an insulated housing similar to the cell and bypass line housings. These housings are maintained at elevated temperatures to prevent condensation of the process gas.

Process piping, with its associated flanges, valves, and expansion joints moves the UF_6 from compressor to converter, cell to cell, and building to building. UF_6 piping is nickel-plated internally to reduce corrosion. Process piping is heated either by routing the piping through heated enclosures or by steam tracing and insulation. The cascade piping requires expansion joints to allow for some pipe misalignment and thermal expansion. The joints are of the monel bellows-type and are designed to operate at 30 psig.

The three main process buildings , X-333, X-330, and X-326, house the equipment and much of the support systems necessary for the isotopic separation of uranium. All process buildings are connected by tie-line piping to form a cascade of the desired throughput capacity.

The ground floor houses the equipment required to provide auxiliary services to the cell floor process equipment. The cell floor (or second floor) houses the stages necessary for the isotopic separation of uranium.

Cells, booster stations, and UF_6 piping are enclosed in a housing of ribbed galvanized sheet metal and transite sheets attached to steel framing. The purpose of the housing is to retain heat so that the UF_6 is maintained in the gaseous state as required by the process. The framing and housing enclosing cells and booster stations are independent of the building framing and are supported by the cell floor.

A steel cylindrical lube oil storage tank is housed approximately at the center of each unit, housed in rooms on the building roof on X-326 and X-333 and near the roof line in X-330. Each cascade unit is provided with a lubricating system to maintain a continuous supply of oil to the compressor and motor bearings. The process recirculating oil system is designed to maintain a continuous and adequate supply of lubricating oil at a controlled temperature and free of abrasive matter to the bearings of axial compressors, centrifugal compressors, and electric motors of the cascade process equipment. The unit lube oil systems also serve the booster pumps and motors as well as hydraulically operated stage control valves in the 000-size equipment.

Individual units within the cascade areas are equipped with a completely independent recirculating lube oil system, which include storage capacity, recirculating pumps, oil strainers, an oil cooler, and a gravity supply tank. This equipment, coupled with the necessary supply, drain and vent piping, and the instrumentation for manual and automatic control, comprises the basic components of the system.

The cell floor area is served by permanent overhead bridge cranes. These cranes are used to handle heavy process equipment during maintenance or modification.

Ionization chamber type UF_6 detection systems are used to monitor selected equipment in the cascade buildings. Detectors are installed at the following locations: cell housings, bypass housings, "B" seals, UF_6 condensers, product withdrawal pumps, UF_6 drain stations, and autoclave areas where UF_6 could conceivably be released. The sensing element is an ionization chamber in which air is made conductive by the use of an alpha emitter. The sensitivity of the ionization chamber is controlled by adjusting the size of the opening over the alpha emitter.

The top cells in the PORTS cascade are called the purge cascade. The purge cascade is located in the X-326 Building. The purge cascade removes light molecular weight gases, allows the withdrawal of product from the cascade, and removes intermediate molecular weight gases from the cascade system. Below the product withdrawal point and a few cells above it, the purge cascade provides the same isotopic enrichment function as other enrichment cascade equipment. Above the product withdrawal point, the function is to separate light molecular gases from the residual UF_6 which remains above product withdrawal point and vent them via the tops purge stack to atmosphere. Both functions are accomplished by using equipment similar to that in the enrichment cascade to effect separation using gaseous diffusion.

The 180 stages in the purge cascade are normally operated as a separate cascade. Nitrogen or lights separation from the residual UF_6 is accomplished primarily in the top four cells while the lower four cells are used primarily for isotopic separation under normal

operations. The remaining middle cells provide for a transition as the UF_6 concentration rapidly drops.

Lights containing only traces of UF_6 will normally be withdrawn from the cascade at the top cell in the purge cascade. The lights pass to the tops booster station and then to the purge rate control system. A portion of the lights is recycled to maintain the lights front while the remainder is passed through one of two parallel banks of Alumina traps then on to the jet system and exhausted to the atmosphere by venting through a stack outside the X-326 Building. The vent stack contains various sampling systems to track stack emissions of fluorides and selected radionuclides.

The Q systems associated with the enrichment operations are: the coolant pressure relief system, criticality accident alarm system, freezer/sublimer high-high weight trip, UF_6 smoke detection systems.

4.3 UF_6 Product Withdrawal

Liquid phase UF_6 withdrawals are performed at three fixed facilities: the X-330 Tails Withdrawal Station, the X-333 Low Assay Withdrawal Station (LAW), and the X-326 ERP Station. Gas phase withdrawals may be performed at the X-326 Product Withdrawal (PW), the X-330 Interim Purge Facility, or at any local cell control panel in the isotopic cascade, using portable equipment.

UF_6 from the diffusion cascade is compressed using centrifugal compressors to a pressure of approximately 30 psia and then cooled to about l60°F to condense the gas. The liquified UF_6 flows by gravity into 30-inch or 48-inch cylinders. Each cylinder being filled is mounted on a scale that monitors the cylinder weight. When the predetermined cylinder weight limit is almost reached, an audible alarm on the scale is sounded to alert the operator, and the valve in the UF_6 drain line to the cylinder automatically closes. The filled cylinder is disconnected and moved outside the building for cooling and solidification of the UF_6 product. Noncondensible contaminant gases remaining in the cylinder are removed by connecting the cylinder to the vent return header and evacuating it.

Two UF_6 liquid accumulators serve the withdrawal loop at the ERP and LAW and Stations. The accumulators' sizes determine the maximum allowable assay that may be withdrawn. The accumulators are constructed of monel-lined steel. The accumulators provide surge volume by "floating" on the drain line. A vent line with a control valve is provided to permit the return of noncondensibles to the cascade and to control pressure.

Each filling station has a cylinder cradle arrangement mounted on a cart, which is moved on a floor track system. A cylinder to be filled is placed in the cradle and the cart is moved into position on a scale at the filling station. A removable pipe or "pigtail" connects the filling station to the cylinder valve. Withdrawal pigtails are wrapped with electrical heat tracing covered with insulating tape to reduce the likelihood of freeze-out.

While the liquid UF_6 is being drained from a condenser or accumulator into the cylinder, the weight of the material withdrawn can be read from the scale to determine when the weight is approaching the cylinder target weight. The cylinder target weight is set at or below the

cylinder fill limit. When this target weight has almost been reached, an alarm on the scale sounds to alert the operator and a valve in the UF_6 drain line automatically closes to prevent overfilling of the cylinder. Additional filling of the cylinder to the target weight is then performed using a local manual override. If the target weight or the cylinder fill limit should be exceeded, the excess material may be removed by valving the cylinder to the evacuation header at the withdrawal position. After being filled, cylinders are moved to the cooldown yard.

The LAW and ERP areas each have one 20-ton overhead bridge crane that is used to move liquid-filled cylinders from the scale cart to the storage area. Cylinder movement from the scale cart to the cylinder cool-down area is minimized and restricted to the lowest practicable height above ground.

Cylinder storage facilities are used for the cool-down of liquid UF_6 cylinders prior to the burping operation. Cylinder saddles are used to position cylinders for storage.

Activation of two UF_6 release detector heads above any drain position pigtail will close the air operated pigtail manifold block valve and the cylinder safety valve.

The Q systems associated with product withdrawal are: ERP and LAW Station UF_6 smoke detection systems, ERP and LAW Station pigtail isolation systems, criticality accident alarm system, ERP and LAW Station UF_6 cylinder pigtails, ERP and LAW Station liquid UF_6 process piping and valves.

4.4 UF_6 Tails Withdrawal

The process for tails withdrawal is similar to that for product withdrawal. The tails withdrawal facility is located in Building X-330.

The Q systems associated with tails withdrawal are: the tails pigtail isolation system, liquid UF_6 handling cranes, liquid UF_6 cylinder lifting fixtures, tails UF_6 cylinder pigtails, tails liquid UF_6 process piping and valves.

4.5 UF_6 Cylinder Storage

Operation of PORTS requires the storage of large inventories of UF_6 in cylinders. This occurs in the form of interim storage facilities for feed and product stockpile and long-term storage for the cylinders containing depleted UF_6 tails. UF_6 cylinder handling equipment is provided for the many operations that must be accomplished.

Large cylinders of the 10-ton and 14-ton size are stored in the X-745B, X-745C, X-745E, X-745F, and X745G outside storage lots and at two additional processing lots (X-343N and X343S) at the X-343 Feed Vaporization Facility. Empty feed cylinders can be stored east of X-343 in a gravel lot. Product cylinders of the 2 1/2-ton size are stored at the X-745B storage lot, which is the processing lot for the X-344 toll Enrichment Facility.

Because of their weight, the cylinders require the use of large specialized handling equipment and a firm foundation for both transport and storage. The cylinders are placed on supports to prevent them from rolling and allow them to be stacked. The saddles also help in corrosion prevention by keeping the cylinders off the ground and as dry as possible. Steel reinforced concrete cylinder saddles are used in newer cylinder yards for cylinder support.

4.6 Chemical Facilities

The X-705 Decontamination and Recovery Facility houses decontamination, uranium recovery, cylinder cleaning and testing, and process laboratory equipment. Other facilities, including processes for F_2, Cl_2, ClF_3, H_2SO_4, and HNO_3, are located within the fenced portion of the site.

The Q systems associated with the chemical facilities are: the calciner high temperature shutoff system, calciner discharge throat level control system, calciner can level control system, microfiltration pH shutdown system, microfiltration permeate effluent bag filter system, and the criticality accident alarm system.

4.7 Conclusions and Compliance Plan Issues

The staff is aware that the descriptions contained in Chapter 3 of the SAR may not match the "as found" condition of the facility in all instances. This aspect is being remedied as part of the SAR upgrade effort. The regulations also provide a mechanism for dealing with "as found" conditions. Chapter 3 of the SAR does provide an adequate description of the facility and process. Therefore, the staff concludes that the facility and process descriptions contained in Chapter 3 of the SAR meet the requirements of the regulations, and are, therefore, acceptable.

There are three Compliance Plan Issues that are associated with: (1) upgrades to the autoclaves, (2) UF_6 leak detector sensitivity testing, and (3) operational trips. These are discussed below.

Issue 3 concerns upgrades to the autoclaves. These upgrades include:
(1) Provide the capability to test the containment valves (i.e., inner and outer loop valves) separately for the autoclaves in X-342A, X-343, and X-344A. This upgrade will be completed and a revised TSR to reflect the new configuration will be submitted to NRC by July 1, 1997; (2) Upgrade the UF_6 feed isolation and flow control valves in X-342A and X-343 to be fail-safe on loss of air or electric power. Upgrade the containment valves on the liquid UF_6 drain line to be fail-safe on loss of air or electric power on autoclaves 3 and 4 in X-344A. Upgrade the daughter cylinder isolation valves on the autoclaves in X-344A to fail close on loss of air; (3) Add a low air pressure switch to autoclave 2 in X-344A in order to initiate containment upon loss of air; (4) Upgrade the High Pressure Containment Shutdown System control logic for the autoclaves in X-342A, X343, and X-344A to have the capability to lock out the hydraulics to prevent the autoclaves from being inadvertently opened when the pressure in the autoclaves exceeds the setpoint; (5) Upgrade the internal autoclave and UF_6 cylinder pressure transmitters to improve their temperature compensation capability and accuracy in the operating pressure range; (6) Provide

operational alarms on the autoclave safety systems to alert operators to potential upset conditions; (7) Modify the autoclave steam supply and condensate removal system in X-342A, X343, and X-344A to minimize the back up of condensate in the autoclave; (8) Restore the autoclave head/shell sealing surfaces for the autoclaves in X-342A, X343, and X-344A; and (9) Obtain a code interpretation from the ASME Code Committee regarding the need for pressure relief for the UF_6 cylinders.

The scheduled completion date for Actions 2 through 9 for the first of thirteen autoclaves is May 1, 1998. The completion schedule for the remaining twelve autoclaves will be such that the final autoclave is complete by February 1, 2001. A detailed schedule for completion of these actions will be available for review at PORTS by March 31, 1998. The staff has reviewed PORT's justification for continued operations and the plan of action and schedule, and based on risk considerations, the staff finds Issue 3 to be acceptable.

Issue 40 addresses operational trips and alarms that may exist which are set to avoid an actual actuation of the associated safety system. PORTS will perform a review of system design where operational trips and alarms coincide with the setpoints for safety system actuation based on the same monitored parameter and the same equipment actuated. The results to this review will be submitted to the NRC for review and approval by October 1, 1996. In the interim, all operational trips and alarms that perform the same function as the associated safety system actuation will be treated and reported as an actuation of a safety system.

Issue 42 addresses the sensitivity of UF_6 leak detector testing. PORTS has not developed detector testing methods which establish a precise correlation between the detectability of "test smoke" and the detectability of UF_6 and its reaction products. Plant experience have demonstrated that the detectors have the capability to detect very small leaks although this small amount has not been quantified. PORTS plans to complete the development of a detector testing methodology by July 31, 1997.

The staff has reviewed the justification for continued operations and the plan of action and schedule for Issues 40 and 42 and finds them to be acceptable.

Chapter 5 ACCIDENT ANALYSIS

The regulations in 10 CFR §76.35(a)(4) require the SAR to include an "assessment of postulated accidents based on the requirements of §76.85." A "reasonable spectrum of postulated accidents which include internal and external events and natural phenomena" is to be considered in the accident analyses. Chapter 4 of the SAR contains the PORTS accident analysis. By TSR 3.20, USEC is required to make changes to the accident analysis in accordance with the plant design change control process described in SAR §6.3. The SAR is based, in part, on the 1985 Final Safety Analysis Report (SAR 85) and approved safety evaluations performed by the plant during the intervening time period. The SAR does not fully incorporate all of the information that is being generated by the current DOE site-wide safety analysis upgrade effort (SAR upgrade). A number of areas and supporting safety analyses are expected to be updated concerning the descriptions of hazards, and plant SSCs and human activities relied on for safety. The SAR upgrade is scheduled for completion by DOE in February 1997 and submittal by USEC to NRC in August 1997 in accordance with Compliance Plan Issue 2.

The potential consequences of postulated accidents include personal injury, health effects from acute exposure to toxic chemicals, non-stochastic effects from acute radiation exposure, and risk of latent cancer because of exposure to radioactive material. The purpose of accident analysis is to (1) investigate the nature and consequences of accidents in order to determine the impact of the facility on the health and safety of the workers and the general public, and (2) identify limiting conditions for operations (LCOs) to assure the safe storage, handling and processing of radioactive and other hazardous materials at PORTS.

5.1 Accident Evaluation Methodology

The staff review included the accident scenario initiators, related release mechanisms, and the potential consequences of each scenario described in Chapter 4 of the SAR with regard to the hazards associated with (1) fire, (2) toxic/radioactive material release, (3) explosion, (4) radiation exposure, and (5) nuclear criticality. The staff considered the facility design and the description of equipment and operations as presented in the SAR and in USEC's responses to staff questions. In addition, the operating history of the three GDPs was also reviewed. As a result, it was determined that the major hazard at a GDP was an accidental release of a large amount of UF_6 or an accidental criticality. Section 1.4 of this CER summarizes the past UF_6 release accidents at the GDPs. Although there have been no criticality events at any of the three GDPs, criticality safety was evaluated.

5.1.1 UF_6 Releases

UF_6 chemically reacts with the moisture in the air to form uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF). The acute toxic chemical effects of exposure to uranium (U) and HF have previously been evaluated by the NRC in NUREG-1391 (NRC, 1991) and by Pacific Northwest Laboratory (PNL) for the NRC in PNL-10065 (PNL, 1994). The NRC analysis concluded that the chemical effects of exposure to uranium in soluble form exceeded the acute radiological effects and that the threshold for renal injury (nephrotoxicity) appears to lie very near 3 micrograms (μg) U per gram kidney tissue which

results from a single inhalation intake of about 30 milligrams (mg) U by a standard person. NRC staff evaluation of the data provided in SAR, Appendix A of Chapter 4, Figure A.1-1, indicates that the thresholds for renal injury are airborne concentrations of 15 mg-U/m^3 for exposure times of one hour or more and 25 mg-U/m^3 for exposure times of 30 minutes or less.

Regarding the radiological hazards of soluble uranium, the equivalent radiological doses for a 30 milligram intake via the inhalation pathway are about 50 millirem for natural uranium, 110 millirem for 2.75% ^{235}U, and 350 millirem for 10% ^{235}U. For the ingestion pathway, the corresponding radiological doses are substantially lower. Therefore, the primary concern for exposure to soluble uranium is from the chemical rather than radiological effects.

According to NUREG-1391, (NRC, 1991) exposure to HF at a concentration of 25 mg/m^3 for 30 minutes has been identified as the level with no significant effects, either short-term or long-term. However, significant irritation of lung tissue could occur at this concentration (PNL, 1994). The American Industrial Hygiene Association (AIHA, 1988) has defined Emergency Response Planning Guidelines (ERPG) as (a) ERPG-1 (4.1 mg-HF/m^3) at which nearly all individuals could be exposed for one hour without experiencing other than mild, transient health effects or objectionable odor; (b) ERPG-2 (16.4 mg-HF/m^3) at which nearly all individuals could be exposed for one hour without developing irreversible or other serious health effects which could impair an individual's ability to take protective action; and (c) ERPG-3 (41 mg-HF/m^3) at which nearly all individuals could be exposed for one hour without developing life-threatening health effects.

The odor threshold for human beings to HF is 0.0333 to 0.1333 mg-HF/m^3. The equivalent stoichiometric half-hour intake of soluble uranium based on the assumption that the UF_6-H_2O reaction goes to completion, would range from 0.06 to 0.24 mg. Similarly, the stoichiometric amount of uranium in a UF_6 release resulting in an airborne concentration of 50 mg-U/m^3 (equivalent to a half hour intake of 30 mg-U) corresponds to an airborne HF concentration of 17 mg-HF/m^3. It should be noted that for relatively long exposures (on the order of tens of minutes) resulting from UF_6 releases, health effects from uranium intakes would be limiting. However, for short exposures (a few minutes or less), health effects from HF would be more limiting. For most of the large UF_6 release scenarios, it is assumed that the exposure time is limited to 30-minutes or the release duration, whichever is shorter, for a member of the public located off-site. For workers located on site, exposure times for most UF_6 release scenarios would likely be shorter than 30 minutes or the release duration, as a direct result of early detection (alarms, sight, smell, etc.), emergency response procedures implementing the "See-and-Flee" policy, and the application of the comprehensive Worker Protection TSR (TSR 3.23).

A description of the dispersion modeling used for PORTS in evaluating the impact of UF_6 releases is presented in Appendix B of SAR Chapter 4. Since there are no residences located within one kilometer of any of the postulated release points, a distance of 1 kilometer was conservatively assumed for reviewing the impact to an off-site resident. For large UF_6 releases of up to 20,000 pounds inside the process buildings, the dispersion modeling conducted for the SAR established UO_2F_2 and HF concentrations in the cell floor

and operating floor of the X-333 Building and release rates from the building to the environment.

In November 1995 and March 1996, the staff obtained draft versions of HGSYSTEM/UF_6, a computer atmospheric dispersion model. The staff used HGSYSTEM/UF_6 to perform independent modeling calculations for several large UF_6 release scenarios, including large releases of liquid UF_6 (14-ton cylinder valve failure) and gaseous UF_6 (seismic event). These calculations were performed to determine the appropriateness of the Justification for Continued Operation presented in the Compliance Plan's SAR upgrade issue for PORTS and to confirm the results of the dispersion analyses described in the SAR. Although the formats of the consequences differed (the SAR presented health effects in graphical form whereas the staff's analyses generated results in terms of uranium airborne concentrations for various distances), the results of the NRC staff's analysis compared reasonably well with the results contained in the SAR. The PORTS SAR upgrade will redo the dispersion analyses using a revised version of HGSYSTEM/UF_6. More detailed confirmatory analyses will be subsequently performed by the NRC staff following of the PORTS SAR upgrade.

5.1.2 Criticality

All criticality pathways and scenarios cannot be identified with a high degree of certainty. Therefore, a reasonable spectrum of criticality events was developed for the X-326 Building processes in the SAR to envelope the likely consequences for all credible events at PORTS. The analysis of those events used the AIREK code and covered a variety of uranium/water mixtures ranging from dry to predominantly aqueous.

Because a special set of conditions involving mass, moderation, and configuration are necessary to achieve a criticality, the SAR analysis consisted of a systematic approach to first identify possible criticality endpoints and then examine the necessary conditions that would allow the hypothesized criticality to occur, whether by accident or normal operating conditions. Those conditions were compared with appropriate nuclear criticality safety criteria and the margin of safety was evaluated. Since it is not always possible to determine the magnitude of the existing safety margin, a number of KENO calculations were performed for realistic accident configurations.

All of the X-326 processes analyzed in the SAR conformed to the double contingency criteria. For those sequences where the contingency relied heavily upon administrative controls, that were not clearly independent, or were not clearly of low probability, the margin to criticality was more closely examined. For most of these cases parametric KENO calculations were performed in the SAR to more closely describe the conditions of mass, moderation, or geometry actually required to achieve criticality, and then the likelihood of occurrence was assessed. No attempt was made in the SAR to estimate the probabilities of accident sequences other than to recognize that the criticality safety criteria should provide assurance that the probabilities of the identified sequences are very small.

Processes for which double contingency criteria cannot be met were identified and the proposed criticality safety controls were evaluated by the NRC staff for inclusion in the TSRs.

Dose estimates for individual accident scenarios in each process building were derived from the formulas contained in Regulatory Guide 3.34, Revision 1, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in Uranium Fuel Fabrication Plants."

5.2 Potential Hazards of Credible Accident Scenarios

Staff review of the PORTS potential UF_6 accident scenarios presented in Chapter 4 of the SAR included evaluation of equipment and procedures against potential failure modes. The review identified and confirmed design features and operating changes which could detect and prevent accidents and response actions which could mitigate the consequences. As a result of this review, the adequacy of several proposed TSRs and application of QA to the safety related SSCs was determined.

The description of process design, equipment, instruments and controls, and administrative controls and safety programs presented in the PORTS SAR provided a basis for reviewing the adequacy of potential accident scenarios. The scenarios presented in the PORTS SAR are for plant segments defined by process function. The plant segments are Feed, Enrichment Cascade, Product Withdrawal, Tails Withdrawal, UF_6 Sampling and Transfer, UF_6 Cylinder Storage, Chemical Facilities, Waste Management and Laboratory Facilities.

5.2.1 UF_6 Releases

Two of the largest representative UF_6 release scenarios (14-ton liquid UF_6 cylinder release and process gas release due to a large seismic event) were independently evaluated by the staff in terms of impacts to an off-site resident. Source terms were obtained from SAR Table 4.9-1 which summarizes 33 accident scenarios.

Assuming reasonably conservative HGSYSTEM/UF_6 input data for a 14-ton (28,000 pound) liquid UF_6 cylinder valve failure scenario (opening at the bottom of the cylinder with a diameter of 1 inch, Pasquill-Gifford atmospheric stability class D (PG-D), and a wind speed of about 5 m/s), an airborne concentration on plume centerline 1 kilometer from the release point, would result in a standard person's 1/2 hour uranium intake of roughly 60 milligrams. For PG-B with a 3 m/s wind speed, an intake reduction of about an order of magnitude was predicted by HGSYSTEM/UF_6 at the same distance.

For a seismically induced UF_6 release of 20,000 pounds, a very conservatively estimated 1/2-hour plume centerline intake (point source, 20 meter stack horizontal release, no building wake effects, PG-F, and 1.5 m/s wind speed) at 1 kilometer was calculated to be roughly 30 mg U. Assumption of an area source (several release points) would significantly reduce this intake since very little plume overlap was observed for PG-F at 1 kilometer even for widely separated release points in a process building.

It should be noted that at short distances (up to several hundred meters) for the cylinder valve failure and seismic scenarios, the calculations showed that severe renal injury or death from exposure to uranium or HF cannot be ruled out. It should also be noted that application of the straight line Gaussian dispersion methodology provided in Regulatory

Guide 1.145 (NRC, 1983) for the seismic scenario, raised intakes at 1 kilometer by almost an order of magnitude.

Although it is clear that the primary chemical hazard present at PORTS is UF_6, SAR 85, and consequently the application, do not include results of a structured hazard audit to identify materials, equipment, and energy sources which could pose a radiological/chemical threat to the health and safety of the worker or public. A detailed, comprehensive hazard audit, as described in Issue 2 of the PORTS Compliance Plan is being conducted. Absent the results of a detailed hazard audit, the areas and buildings where the significant quantities of hazardous material exist were generally determined based on the information contained in the PORTS SAR. This information is summarized below for the major plant facilities where a significant hazard exists to a member of the public located off site, a worker located on site or to the environment.

Enrichment Cascade Systems (X-333, X-330 and X-326 up to the UF_6 Withdrawal Pump Suction)

Large quantities of UF_6 are processed in the diffusion cascade, located in Buildings X-333, X-330 and X-326. Operating at the maximum (CUP) power of 2260 megawatts (MW), X-333 would contain in process, roughly 775,000 pounds of gaseous UF_6 within 640 of the largest stages and within process piping and equipment. Building X-330 would contain roughly 275,000 pounds of gaseous UF_6 in 1100 of the relatively smaller stages, and in process piping and equipment. Building X-326 would contain only about 3,000 pounds of gaseous UF_6, since the stage sizes are comparatively the smallest and most of the enrichment cascade equipment in X-326 has been shut down. It should be noted that at CUP power, even though most of the Process Gas (PG) Hi-side (compressor exhaust) pressures of the large stages exceed 14.5 psia (atmospheric pressure), most of the cascade is operated at sub-atmospheric pressures and therefore most PG confinement failures even at CUP power would initially result in ambient air or seal buffer gases leaking into the cascade as opposed to PG leaking out. In addition, USEC is committed to application of NQA-1 QA requirements in a graded manner for controlling the quality of piping and equipment containing large quantities of UF_6 and maintaining the PG pressure (Hi-side compressor exhaust pressure) below 25 psia for uprated equipment and 14.5 psia for non-uprated equipment which were the cascade test pressures during the cascade improvement and upgrade project (CIP/CUP) of the mid-1980s. This pressure limitation also ensures that the pressure of a cell operating near 25 psia would drop to, or below, atmospheric pressure soon after the cell is tripped, terminating the initial driving force for any release. The cascade system is described in Chapter 4 of this CER.

The Cascade segment of the plant includes piping and equipment (compressors, converters, valves, heat exchangers, etc.). The Staff reviewed credible UF_6 release accident scenarios pertaining to the Cascade segment of the plant as presented in Chapter 4 of the PORTS SAR. These scenarios and their bounding UF_6 release quantities inside the cascade building (in gaseous form unless specified) are listed below.

HIGH TEMPERATURE ACCIDENT SCENARIOS

(1)	Scenario R-1a	Failure of Cell Cooling Due to RCW Failure - 10 Lbs
(2)	Scenario R-1b	Failure of Cell Cooling Due to Removal of R-114 - 10 Lbs
(3)	Scenario R-2a	Compressor Failure (Deblade) Followed by UF_6/Al Exothermic Reaction - 1,000 Lbs
(4)	Scenario R-2b	Compressor Failure (Deblade) Due to Closure of Recycle Valve on Off-Stream Cells - 10,000 Lbs

HIGH PRESSURE ACCIDENT SCENARIOS

(5)	Scenario R-3	Inadvertent B-Stream Block Valve Closure - 10,000 Lbs
(6)	Scenario R-4	Stage Control Valve Closure Due to Rupture of B-Stream Instrument Line - 10,000 Lbs
(7)	Scenario R-5	Stage Control Valve Closure Due to UF_6 Freezeout in Instrument Line - 10,000 Lbs

EXPLOSION ACCIDENT SCENARIO

(8)	Scenario R-6	Explosive Condition Created by Use of Cell Treatment Gas (ClF_3, F_2) - 10 Lbs

OPERATOR ERROR ACCIDENT SCENARIOS

(9)	Scenario R-7	Heavy Equipment Drop From Crane on Process Piping - 4,000 Lbs
(10)	Scenario R-8	Improper Purging of Cells Devoid of Most UF_6 - 10 Lbs
(11)	Scenario R-9	Process Piping Rupture Caused by Fork Lift - 6,000 Lbs
(12)	Scenario R-10	Improper Evacuation of Cells Containing UF_6 - 2,400 Lbs/Hr

EQUIPMENT/PROCESS PIPING FAILURE ACCIDENT SCENARIOS

(13)	Scenario R-11	Motor to Compressor Coupling Failure Resulting in Rupture of Process Piping - 21,250 Lbs
(14)	Scenario R-12	Process Piping Expansion Joint or Weld Joint Fatigue Failure From Vibration or High Temperatures - 1,000 Lbs
(15)	Scenario R-13	Compressor Seal Failure - 10 Lbs

(16) Scenario R-14 Process Piping Expansion Joint or Weld Joint Failure Due to a 250-Year Seismic Event - 20,000 Lbs

PURGE CASCADE ACCIDENT SCENARIOS

(17) Scenario R-15 Inadvertent Overpressurization of Cell with N_2 Resulting in Seal Leak - 1 Lb

(18) Scenario R-16 Failure to Obtain UF_6 Negative During Cell Purge - 1 Lb

(19) Scenario R-17 Misvalving at Booster Station Resulting in Outdoor Release Through Air Ejectors - 0.25 Lbs/min

FREEZER SUBLIMER ACCIDENT SCENARIO

(20) Scenario R-18 Stress Rupture of Freezer Sublimer Vessel Due to UF_6 Bridging Between Fins/R-114 Tubes, or Heating Overfilled Vessel - 10 Lbs

COLD RECOVERY ACCIDENT SCENARIOS

(21) Scenario R-19 Rupture of Surge Drum Expansion Joint Due to Overpressurization Using 100 psig Plant Air - 250 Lbs

(22) Scenario R-20 Misvalving During Cold Trap Operation Resulting in Outdoor Roof Release Through Jet Exhauster - 2 Lbs/min

(23) Scenario R-21 ClF_3/R-114 Explosion in Cold Trap - 160 Lbs Solid

(24) Scenario R-22 Inadvertent Routing of ClF_3 to Alumina Traps Resulting in Outdoor Release of Trapped UF_6 - 10 Lbs

(25) Scenario R-23 Fire Caused by Overheating of Cold Trap Resulting in Rupture - 2 Lbs

(26) Scenario R-24 Misvalving and Pressure Blind Switch Failure During Cold Trap Operation Resulting in Overpressurization Rupture - 160 Lbs Solid

(27) Scenario R-25 Condensation and Subsequent Explosion Caused by ClF_3 Reaction Products in Cold Trap - 160 Lbs Solid

(28) Scenario R-26 Misvalving in Returning Holding Drum Contents to Cascade Resulting in Outdoor Roof Release Through Jet Exhauster - 40 Lbs/min

The primary controls in the form of technical safety requirements (see CER Chapter 6) and contained in SAR Chapters 3 and 4 that are relied upon to prevent or mitigate significant

potential accident scenarios in the Cascade segment of the plant, and to ensure that the bounding source term is not exceeded are:

(1) Cell Trip System

(2) UF_6 Pressure Limits

(3) Coolant Pressure Safety Limit and High Pressure Relief System

(4) High Pressure Fire Water System

(5) Cell Treatment and Other Explosive Gas Monitoring and Explosion Prevention Administrative Requirements

(6) Crane Movement of Cascade Equipment Administrative Requirements

(7) Freezer/Sublimer Weight Safety Limitations, High Weight Cascade Trip System, and Venting Requirements

(8) Prohibition on Applying Direct Heat to UF_6 Solidified Plugs in Cascade Equipment

(8) Cold Trap Pressure Relief System

(9) CADP UF_6 (Smoke) Detection System

(10) UF_6 Piping and Equipment (diameter greater than 2 inches), UF_6 Confinement and QA Requirements

(11) Administrative Controls Specifically Identified in the TSRs and Contained in Programs and Procedures Required by Section 3 of the TSRs

The staff has reviewed the accident cascade scenario descriptions and the associated controls presented in the submitted PORTS TSRs, including the TSR Basis Sections, and PORTS SAR Section 3.8. Based on the information provided in the USEC application, the staff has determined that the scenarios described appear to constitute a reasonable spectrum of postulated accidents and that the safety controls for preventing significant UF_6 releases are adequate. In addition, the staff has reviewed and found acceptable the safety rationale contained in the Compliance Plan issues dealing with the cascade segment of the plant for justifying continued operation.

UF_6 Feed, Withdrawal, Transfer and Sampling Facilities (X-342, X-343, X-344, ERP, LAW, Tails)

In the Feed, Transfer (Toll Enrichment) and Sampling facilities, the major pieces of equipment identified as potentially hazardous because of the presence of liquid UF_6 and thermal energy, are the feed, transfer and sampling autoclaves. There are 13 autoclaves at PORTS. Cylinders containing as much as 14 tons of UF_6 are heated by steam in these autoclaves to liquify the solid UF_6. In addition to being able to provide secondary

confinement, the autoclaves have several appropriate safety systems to prevent or mitigate accidental releases of UF_6. The autoclaves and their associated safety systems and operations are described in Chapter 4 of this CER.

The Feed segment of the plant (X-342 and X-343) includes steam heated feed autoclaves, and associated valves, piping, and controls. The significant potential accident scenarios inside autoclaves include failure of cylinder valves and pigtails, hydraulic rupture from overheating a UF_6 cylinder, or heating an overfilled UF_6 cylinder, and cylinder rupture caused by an explosive hydrocarbon-UF_6 reaction. The significant potential accident scenarios outside autoclaves in the feed facilities include mechanical failure of a cylinder (e.g., cylinder drop), and failure of UF_6 cascade feed piping.

The Toll Enrichment and Sampling segment of the plant (X-343 and X-344) includes steam heated sampling and transfer autoclaves, and associated valves, piping, and controls. The significant potential accident scenarios inside autoclaves include failure of cylinder valves and pigtails, hydraulic rupture due to overheating of a UF_6 cylinder or heating of an overfilled UF_6 cylinder, and rupture due to hydrocarbon-UF_6 reaction. The significant potential accident scenarios outside or in open autoclaves include mechanical failure of a liquid UF_6 cylinder (e.g., cylinder drop), failure of cylinder valves and pigtails, and failure of the UF_6 transfer and sampling piping.

In the Product and Tails withdrawal facilities (ERP, LAW and Tails), the major hazard is UF_6 which exists above atmospheric pressure mostly as liquid in cylinders (up to 14 tons), and in process equipment and piping. Compressors withdraw PG (tails and product) from the cascade raising the pressure above 1 atmosphere. The gaseous UF_6 is cooled and liquified before being withdrawn into product and tails cylinders. Product and Tails withdrawal operations are described in Chapter 4 of this CER.

The Product and Tails withdrawal segments of the plant include UF_6 withdrawal compressors, condensers, accumulators and associated valves, piping, cylinders, and controls. The significant potential accident scenarios in the Product and Tails withdrawal segments include mechanical failure of UF_6 cylinders, failure of cylinder valves and pigtails, hydrocarbon-UF_6 reaction, and failure of UF_6 confinement (compressor to UF_6 withdrawal manifold including UF_6 accumulator and condenser).

The staff reviewed credible UF_6 release accident scenarios pertaining to the UF_6 Feed, Withdrawal, Transfer, Sampling, and Cylinder Handling operations at the plant as presented in Chapter 4 of the PORTS SAR. These scenarios and their bounding UF_6 release quantities (in gaseous form unless specified) are listed below.

FEED, WITHDRAWAL, CYLINDER TRANSFER AND CYLINDER HANDLING ACCIDENT SCENARIOS

(1)	Scenario R-27	14-Ton UF_6 Cylinder Rupture - 28,000 Lbs Liquid
(2)	Scenario R-28	UF_6 Cylinder Pigtail Rupture - 14-Ton 28,000 Lbs, 10-Ton 20,000 Lbs Liquid

(3)	Scenario R-29	UF_6 Cylinder Valve Failure - 6 O'Clock Release 28,000 Lbs Liquid, 12 O'Clock Release 720 Lbs
(4)	Scenario R-30	Explosion Caused by Liquid UF_6/Hydrocarbon Oil Reaction During Withdrawal and UF_6 Cylinder Heating - 28,000 Lbs Liquid
(5)	Scenario R-31	Compression Loop Piping (Expansion Joint) Fatigue Failure - 1,000 Lbs
(6)	§4.2.3.3	Condenser/Accumulator/Withdrawal Manifold Piping Failure - ERP and LAW 500 Lbs Liquid, Tails 2,000 Lbs Liquid
(7)	§4.2.3.4	Pigtail Gasket Failure - 127.5 Lbs Liquid
(8)	§4.2.3.5	Buffered Expansion Joint Single Bellows Wall Failure - 0 Lbs
(9)	§4.2.3.6	Buffered G-17 Valve Failure - 0 Lbs
(10)	§4.2.3.7	Compressor Seal Failure - ERP and LAW 500 Lbs, Tails 2,000 Lbs
(11)	§4.2.3.9	Tails Withdrawal Header and Surge Drum Failure - 0 Lbs
(12)	§4.2.3.10	Pigtail Connection Leaks - Several Grams
(13)	Scenario R-36	Side Withdrawal Piping Rupture Caused by Mobile Equipment - 10 Lbs From Cell

USEC has identified primary controls in the form of technical safety requirements (see CER Chapter 6 and SAR Chapters 3 and 4) that are relied upon to prevent or mitigate significant potential accident scenarios at the Feed, Toll Enrichment and Sampling segments of the plant. These are summarized below.

(1) Autoclave Shell Pressure Limits

(2) UF_6 Cylinder Temperature Limits

(3) UF_6 Cylinder High Pressure Autoclave Steam Shutoff System

(4) UF_6 Cylinder Low Pressure Autoclave Steam Shutoff System

(5) UF_6 Cylinder High Temperature Autoclave Steam Shutoff System

(6) Autoclave High Condensate Level Steam Shutoff System

(7) Autoclave Shell High Steam Pressure Steam Shutoff System

(8) Autoclave Shell High Pressure Containment System

(9) Autoclave Shell High Pressure Relief System

(10) UF_6 Cylinder Filling and Heating Limitations

(11) Autoclave UF_6 (Smoke) Detection System

(12) UF_6 Cylinder Pigtail Isolation System

(13) Prohibition on applying direct heat to solidified UF_6 plugs in piping and equipment

(14) UF_6 Cylinder, UF_6 Cylinder Overhead Crane and Lifting Fixture, Pigtail, and Raschig Ring Design and Quality Control Requirements

(15) Liquid UF_6 Cylinder Lifting and Movement Restrictions

(16) UF_6 Cylinder, Pigtail, UF_6 Piping (greater than 2 inches in diameter), and Autoclave Quality Control and Confinement Requirements

(17) Administrative controls specifically identified in the proposed TSRs and contained in programs and procedures generically required by Section 3 of the proposed TSRs

The primary controls in the form of technical safety requirements (see CER Chapter 6) and contained in SAR Chapters 3 and 4 that are relied upon to prevent or mitigate significant potential accident scenarios at the Product and Tails Withdrawal segments of the plant are:

(18) UF_6 Cylinder Pressure Limits

(19) Coolant (R-114) High Pressure Relief System

(20) UF_6 Cylinder Pigtail Line Isolation System

(21) UF_6 Cylinder Fill Weight, Heating and Movement Limitations

(22) Liquid UF_6 Cylinder Handling, Crane Operability Requirements

(23) UF_6 (Smoke) Detection System

(24) Prohibition on applying direct heat to solidified UF_6 plugs in piping and equipment

(25) UF_6 Cylinder, UF_6 Cylinder Overhead Crane and Lifting Fixture, Pigtail, and Raschig Ring Design and Quality Control Requirements

(26) UF_6 Cylinder, Pigtail, Accumulator, Condenser, UF_6 Piping, Compressor and Valve Quality Control and Confinement Requirements

(27) Administrative controls specifically identified in proposed TSRs and contained in programs and procedures required by Section 3 of the proposed TSRs

The staff has reviewed the non-cascade accident scenario descriptions and the associated primary controls presented in the submitted PORTS TSRs, including the TSR Basis Sections, and PORTS SAR Chapters 3 and 4. Based on the information provided in the USEC application, the staff has determined that the scenarios described appear to constitute a reasonable spectrum of postulated accidents and that the safety controls for preventing significant UF_6 releases are adequate. The staff has also reviewed and found acceptable the safety rationale contained in the Compliance Plan issues for justifying continued operation.

OTHER FACILITY ACCIDENT SCENARIOS

Other PORTS facilities that introduce additional risk to on-site worker health and safety because of possible accidents resulting in unconfinement of radioactive material include the Decontamination Building (X-705) and the Radiation Calibration Laboratory (X-710). The potential consequences include personal injury, health effects from acute exposure to toxic chemicals, non-stochastic effects from acute radiation exposure, and risk of latent cancer because of exposure to radioactive material. The credible accident scenarios presented in the SAR and their bounding releases, where available, are listed below.

(1)	Scenario R-37	X-345 SNM Solid UF_6 Cylinder Valve Leak - Minor
(2)	Scenario R-37a	Release of Toxic Material in X-705 Affecting X-345
(3)	Scenario R-37b	Release of Toxic Material in X-330, X-333 and X-344 Affecting X-345
(4)	§4.3.1.1.1	Accident Initiating Events Related to X-705: Tornado - 200 Lbs Solid; Flood - 0 Lbs; 0.05g Earthquake - 0 Lbs: Vehicle Impacts - Minor; Explosion - Not Credible; Fire - Not Credible; Corrosive Materials - Localized Minor Impact
(5)	§4.3.1.2	X-705 "A" Area Accidents: Cylinder Cleaning - Minor; Small Parts Cleaning - Minor
(6)	§4.3.1.3	X-705 "B" Area Accidents: Glove Box Fire - 1.3 Rem to Bone in 15 Minutes at 1700 Feet (25 mCi U)
(7)	§4.3.1.4	X-705 "C" Area Accidents: Inadvertent Pressuring Cylinder with Plant Air in South Annex - 13.3 Lbs; Vehicular Impact Resulting in Solid UF_6 Cylinder Valve Breakage - 0 Lbs
(8)	§4.5.1.6	X-710 Accidental Direct Radiation Overexposure; Fire; Toxic Material Release; Radioactive Sample (1S, 2S, U-Tube) Container Releases

The primary controls to prevent or mitigate significant potential accident scenarios at the Decontamination segment of the plant that have been identified by USEC as being relied upon for safety are:

(1) Calciner High Temperature Safety Limit of 2,000 °F

(2) Microfiltration Permeate Effluent Bag Filter High Pressure Differential Safety Limit of 30 psid and Shutdown System

(3) Calciner High Temperature Shutoff System

(4) Administrative controls specifically identified in the proposed TSRs and contained in programs and procedures required by Section 3 of the proposed TSRs

The staff has reviewed the above mentioned accident scenario descriptions and associated controls presented in the submitted PORTS TSRs, including the TSR Basis Sections. Based on the information provided in the USEC application, the staff has determined that the scenarios described appear to constitute a reasonable spectrum of accidents and that the safety controls for preventing significant UF_6 releases are adequate.

5.2.2 Criticality

The following criticality accident scenarios pertaining to the Cascade segment of the plant were analyzed as presented in Chapter 4 of the PORTS SAR.

Enrichment Cascade Systems (X-333, X-330 and X-326)

<u>Accumulation of Solid Masses of Uranium Compounds</u>

(1)	Scenario C-1	A-Line Cooled Cell - Top Stage Freeze-Out
(2)	Scenario C-2	Lube Oil Inleakage through the Labyrinths of Compressor Bearings and Process Seals.
(3)	Scenario C-3	UF_6 Freeze-out in Piping Elbows and B-Line Drops
(4)	Scenario C-4	UF_6 Freeze-out in the Building Tie-Lines
(5)	Scenario C-5	Wet-Air Inleakage
(6)	Scenario C-6	Prolonged Power Failure
(7)	Scenario C-7	UF_6 Freeze-out in Intermediate and Bottom Surge Drums
(8)	Scenario C-8	Water Control Valve Leak and Cell Block Valve Leak Through the Seats on a Shutdown Cell
(9)	Scenario C-9	UF_6 Freeze-out in the Interbuilding Booster Stations

(10) Scenario C-10 Equipment Removed for Maintenance

(11) Scenario C-11 Solid Mass in the X-330 or X-333 Evacuation Booster Stations (EBS)

(12) Scenario C-12 Stage Compressor Vibration and Expansion Joint Rupture

(13) Scenario C-13 Exothermic Reactions Resulting in Uranium Compound Deposits

Solid Uranium Mass Accumulation

(14) Scenario C-14 Dry Solid Mass of UF_6

(15) Scenario C-15 Dry Mass UO_2F_2

(16) Scenario C-16 Moderated Solid Mass (UF_6 or UO_2F_2)

(17) Scenario C-17 Criticality in X-333 8-inch Cold Traps

(18) Scenario C-18 Criticality in X-330 5-inch Cold Traps

(19) Scenario C-19 Criticality in X-333 24-inch Alumina Trap

(20) Scenario C-20 Criticality in X-330 or X-333 Surge Drums

(21) Scenario C-21 Criticality in X-330 or X-333 Holding Drums

UF_6 Feed, Withdrawal, Sampling, Handling, and Cylinder Storage Facilities and Systems

(22) Scenario C-22 Criticality at Product Withdrawals and HASA

(23) Scenario C-23 Criticality in X-344A Due to Improper Cylinder Handling

(24) Scenario C-24 Criticality from Accidental Geometry Change

- Criticality in X-345 SNM Storage Facility

Scenario C-24a Criticality from Water Intrusion

Scenario C-24b Delta Barrier Accident

- Criticality in Buildings Surrounding X-345

Scenario C-24c Criticality in X-330 Building

Scenario C-24d Criticality in X-333 Building

Scenario C-24e Burst in X-705 Building

Scenario C-24f Low Power Reaction at X-705 Building

Uranium recovery and Chemical Systems

Scenario C-25 Criticality at Uranium Recovery Calciners

USEC's criticality program (see Chapter 8) includes controls to prevent criticality accidents. The staff finds that the criticality accident analysis and the criticality safety program, in combination with the Compliance Plan, provide adequate assurance of safety and are acceptable.

Chapter 6 TECHNICAL SAFETY REQUIREMENTS

The regulations at 10 CFR §76.35(e) require the application to contain "Technical Safety Requirements in accordance with 10 CFR §76.87." The TSRs are to consider the information from the safety analysis report. TSRs are to include safety limits (SL), limiting control settings (LCS), limiting conditions for operations (LCO), design features (DF), surveillance requirements (SR), and administrative controls as appropriate.

The staff reviewed the proposed TSRs against the application, the 1985 SAR, the current Operational Safety Requirements (OSRs), safety controls incorporated into existing procedures, nuclear criticality safety requirements for ongoing activities, and current plant safety practices. The staff determined that the TSRs incorporate the safety requirements of the current OSRs and other safety controls utilized by the DOE to ensure safe GDP operations. Several OSR requirements were also clarified during transition to TSRs. TSRs defining administrative or programmatic requirements were consistent with other similar NRC requirements for nuclear power plants or current DOE mandates (DOE Orders and Standards on TSRs) as a part of the Regulatory Oversight Agreement.

PORTS TSRs consist of facility/equipment specific TSRs and administrative TSRs. Section 1 contains acceptable TSR related definitions, surveillance time intervals, acronyms, intent of the "shall," "should," and "may" terms, applicability of SLs, LCSs, LCOs and SRs, and allowable conditions outside the scope of TSRs. Section 2 contains the facility-specific TSRs, including TSRs on the autoclaves, UF_6 detection systems, criticality accident alarm systems, cylinder handling, cylinder filling, cylinder heating, fire protection system, and other process-related equipment. Section 3 contains the TSRs related to administrative controls, including responsibility assignment, the organization, staffing, the PORC, procedures, training, criticality safety, and commitments to the radiation protection, fire protection, chemical safety, environmental protection, radioactive waste management, and maintenance programs, as well as the other topics required by the regulations. The administrative control and programmatic TSRs are discussed in other sections of the CER; a summary of the facility specific TSRs contained in Section 2 is as follows.

Section 2.1 contains the TSRs for the UF_6 feed, sampling and transfer autoclave facilities (X-342, X-343, and X-344). USEC has established operational modes (TSR 2.1.1) that are appropriate for the operations that occur in the UF_6 feed, sampling and transfer facilities. Safety limits for the autoclave shell pressure have been set at 110% of the autoclave MAWPs (TSR 2.1.2.1). Temperature safety limits for UF_6 cylinders (TSR 2.1.2.2) have been set to ensure proper ullage (space inside a cylinder into which liquid UF_6 may expand when heated) for the particular cylinder types routinely in use at the facility. These safety limits are appropriate and provide adequate margins of safety.

TSR 2.1.3.1a requires the detection capabilities of the criticality accident alarm systems to be operable in X-342, X-343 and X-344 Buildings, equipment or processes, which contain greater than 700 grams of ^{235}U enriched to 1.0 weight % ^{235}U or more. TSR 2.1.3.1.b requires the alarming capabilities of the criticality accident alarm systems to be operable in areas where the maximum foreseeable absorbed dose in free air due to criticality would exceed 12 rad. Loss of detection or alarming capabilities requires immediate discontinuation of movement of uranium and cylinders with UF_6, and cascade feeding of

UF_6 enriched to 1.0 weight % ^{235}U or more. However, the current or ongoing transfer and/or sampling operation is permitted to be completed to avoid a competing risk that may be introduced by immediately terminating transfer and/or sampling operations. In addition, only personnel equipped with alternate means of alarm notification are allowed in the area for which coverage has been lost. Restoration of criticality detection and alarming capabilities is required prior to reinitiating activities required to be monitored for criticality. Appropriate potentially abnormal conditions, required actions, and surveillance requirements are established.

The UF_6 cylinder high pressure autoclave steam shutoff systems cause the autoclave steam supply valves to close when the pressure exceeds the appropriate cylinder pressure set point. The SR of TSR 2.1.3.2 identifies actuation pressures of 115 psia (5 psia tolerance) and 16.9 psia (3.1 psia tolerance) for normal (Type A and B) and controlled (cold) fed (Type C) cylinders, respectively. These are appropriate since the MAWPs for thin and thick walled cylinders are 100 and 200 psig, respectively, and during controlled feeding at 20 psia (16.9 + 3.1), liquefaction of UF_6 would not occur. The TSR sets appropriate potentially abnormal conditions, required actions, and surveillance requirements.

In addition to the UF_6 cylinder high pressure autoclave steam shutoff systems, the UF_6 cylinder high temperature autoclave steam shutoff systems would also cause the autoclave steam supply valves to close when the temperatures exceed appropriate cylinder temperature set points. TSR 2.1.3.3 and its SRs identify actuation temperatures (LCSs) of 235 °F, 220 °F, and 145 °F for cylinder categories A, B, and C, respectively. For cylinder categories A and B, the LCS's prevent the cylinders from going solid (i.e. being completely occupied by liquid UF_6) by maintaining an adequate ullage. For cylinder category C, (controlled feeding), the LCS of 145 °F prevents any liquefaction of UF_6 in the cylinders. The TSR sets appropriate potentially abnormal conditions, required actions, and surveillance requirements.

The autoclave shell steam high pressure shutdown system provides an indirect means of controlling cylinder temperature below the safety limits such that a loss of ullage and cylinder overpressurization could not occur. TSR 2.1.3.4 establishes appropriate LCSs of 8.0 and 2.5 psig on the autoclave steam pressure used for heating Category A and B cylinders, respectively, and requires the system to be operable in the (1) Heating, (2) Feeding, Transfer or Sampling, and (3) Controlled Feeding operational modes. It should be noted that unlike its applicability statement, this TSR is not appropriate for controlled feeding since the LCSs only apply to cylinder categories A and B. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established in the TSRs.

The autoclave shell high pressure containment shutdown systems cause the autoclaves to go into containment if a pressure increase indicates a UF_6 release inside the autoclave during the (1) Heating, (2) Feeding, Transfer or Sampling, and (3) Controlled Feeding operational modes. The SR of TSR 2.1.3.5 identifies the LCS as being an actuation pressure of 15 psig. The action statement also requires the autoclave to be placed in the "Out of Service" mode if steam is observed leaking around the autoclave locking ring. The TSR sets appropriate potentially abnormal conditions, required actions, and surveillance

requirements. One of the surveillance requirements is to conduct quarterly autoclave pressure decay or leak rate tests at accident pressures of at least 90 psig.

The autoclave shell high pressure relief system (TSR 2.1.3.6) shall be operable in the (1) Heating, (2) Feeding, Transfer or Sampling, and (3) Controlled Feeding operational modes, and shall have the actuation pressure (LCS) set at a level to prevent the rupture of the autoclave. The LCS are set appropriately at 100% of the autoclave MAWPs with an allowable tolerance of 5%. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established. The autoclave high condensate level shutoff system (TSR 2.1.3.7) shall also be operable to prevent upon a UF_6 release the possibility of (1) pressurization of the autoclave beyond the minimum accident test pressure of 90 psig and (2) a criticality event. Appropriate potentially abnormal conditions, required actions, and surveillance requirements are established in TSR 2.1.3.7.

TSRs 2.1.3.15 and 2.5.3.11 require UF_6 cylinder fill weights to be less than or equal to the listed quantities. In addition, prior to heating, the LCO in TSR 2.1.3.8 requires UF_6 cylinder accountability weights to be less than or equal to the cylinder standard fill weights provided in TSR 2.1.3.15. Minimum ullages for tails cylinders have been set at 3% and for product and feed cylinders at 5%. Except for low specific activity (depleted uranium tails) cylinders, the fill limits contained in TSRs 2.1.3.15 and 2.5.3.11 are consistent with the values included in American Nuclear Standards Institute (ANSI) publication ANSI N14.1-1990 entitled "For Nuclear Materials - Uranium Hexafluoride - Packaging and Transport." It should be noted that for off-site transportation of tails cylinders, ANSI N14.1-1990 permits higher fill limits provided a minimum 5% ullage is assured at a cylinder temperature of 235 °F for a minimum UF_6 purity of 99.5%. Since tails cylinders are not transported off-site from PORTS, they are not subject to transportation requirements, and the minimum requirement of 3% ullage for on-site storage is adequate. TSR 2.1.3.8 includes appropriate calibration and functional testing surveillance requirements for accountability scales.

The UF_6 cylinder low pressure autoclave steam shutoff systems cause the autoclave steam supply valves to close when the pressure remains less than 20 psia after the first hour of heating, indicating a possible closed or plugged cylinder valve. TSR 2.1.3.9 sets appropriate potentially abnormal conditions, required actions, and surveillance requirements. TSR 2.1.3.10 adequately prevents the heating of cylinders containing uranium above 5 weight % in autoclaves by requiring verification that the assay is less than or equal to 5 weight % ^{235}U prior to entering the cylinder heating mode. TSR 2.7.3.10 prohibits the heating of 5, 8 and 12 inch cylinders by a heater rated at more than 500 watts which ensures that UF_6 contained in these cylinders will not be liquified.

According to the LCO of TSR 2.1.3.11, in the autoclave areas of X-344 and X-342, two or more UF_6 release (smoke) detectors and in the autoclave area of X-343, four or more UF_6 release (smoke) detectors shall be operable in all modes of operation. The detection system alarms upon detection of UF_6. This TSR establishes appropriate potentially abnormal conditions, required actions, and surveillance requirements which include smoke testing the detectors semiannually. TSR 2.1.3.13 has been established to isolate the pigtail or the sample/transfer manifold thus mitigating an accidental UF_6 release from the pigtail or the sample/transfer manifold during the (1) Heating, (2) Cylinder/Pigtail Operations, (3) Feeding, Transfer or Sampling, (4) Cold Feeding, and (5) Controlled Feeding

modes of operation. In the event of a UF_6 release detected and alarmed by the UF_6 release detection system, an operator would manually actuate a remote pushbutton to initiate closure of the autoclave isolation valves. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established.

The LCO in TSR 2.1.3.12 requires cranes to be operable prior to lifting liquid UF_6 cylinders. Appropriate potentially abnormal conditions, required actions, and surveillance requirements, which include hands-on testing of the cranes per Office of Safety and Health Administration (OSHA) requirements, have been established. TSR 2.1.3.14 only allows movement of liquid UF_6 cylinders with overhead cranes, scale carts and railcars.

Section 2.2 contains the TSRs for the enrichment cascades in X-330 and X-333 and Section 2.7 contains the TSRs for the enrichment cascades in X-326. USEC has established appropriate operational modes in TSRs 2.2.1 and 2.7.1. Appropriate SLs (110% of the MAWP) have been established for the UF_6 gas coolant (R-114) overpressure protection systems (2.2.2.1, 2.2.3.1 and 2.7.2.1). Overpressurization of the coolant system could force R-114 into the UF_6 system and threaten the UF_6 pressure boundary. These systems also have appropriate LCOs, LCSs (at MAWP), potentially abnormal conditions, required actions, and surveillance requirements (2.2.3.1 and 2.7.3.1).

USEC has established TSRs 2.2.3.13 and 2.7.3.12 to limit in the (1) Operating and (2) Floating modes of cascade operations, the cell high side pressure to less than or equal to 25 psia (test pressure) for most X-31 and all X-33 (larger sized uprated equipment) cells and 14.45 psia for some X-31 and all X-29, X-27, and X-25 (smaller sized non-uprated equipment) cells to prevent rupture of the cascade containment and subsequent UF_6 release. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established. Safety limits are set at 40 psia for uprated and 16 psia for non-uprated equipment by TSRs 2.2.2.2 and 2.7.2.2. Until the SAR upgrade analysis is completed, SLs of 40 and 16 psia are acceptable based on the justification provided by USEC in the TSR 2.2.2.2 Basis, and on the information provided in the Paducah (PGDP) SAR Table 4.3-1. It should be noted that large uprated cascade equipment used at PGDP have similar design features as those used at PORTS. In addition, the SAR states that failure of a compressor shaft seal, which for PGDP is designed to withstand a UF_6 pressure of up to 28 psia and was tested to 60 psia, would only result in a 5-10 pound UF_6 release. Also, absent an acceptable basis provided by USEC and until the SAR upgrade analysis is completed, it appears reasonable to apply a SL of 16 psia (110% of the LCO) to non-uprated cascade equipment since the cell UF_6 inventories are significantly smaller than those for uprated equipment, amounting to a reduced potential risk. TSRs 2.2.3.14 and 2.7.3.13 require the DC control power and air pressure required for tripping cells to be operable in the (1) Operating and (2) Standby operational modes. Operator initiated cell trips followed by cell isolation during accident conditions would mitigate or terminate any UF_6 releases. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established.

TSRs 2.2.3.1a and 2.7.3.2a require the detection capabilities of the criticality accident alarm systems to be operable in the cascade buildings when equipment or processes contain greater than 700 grams of ^{235}U at an enrichment greater than or equal to 1.0 weight % ^{235}U. TSRs 2.2.3.1.b and 2.7.3.1b require the alarming capabilities of the

criticality accident alarm systems to be operable in areas where the maximum foreseeable absorbed dose in free air due to criticality would exceed 12 rad. Appropriate immediate compensatory actions are required on loss of detection or alarming capabilities. During system inoperability, cascade enrichment operations are permitted to continue to avoid a competing risk that could be introduced if enrichment operations in all affected cells are terminated. Appropriate compensatory measures are included for the period of inoperability. Restoration of criticality detection and alarming capabilities is required within 48 hours. Appropriate surveillance requirements are established.

The UF_6 release (smoke) detection system is required to be operable for cells in the Operating mode with a high side pressure (B-Line compressor discharge) above atmospheric pressure (TSR 2.2.3.3). Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established.

USEC has established TSRs for moderation control related to uranium deposits in cascade piping and equipment. TSRs 2.2.3.15, 2.6.3.10 and 2.7.3.14 establish appropriate abnormal conditions, required actions, and surveillance requirements. USEC has also established TSRs 2.2.3.8 and 2.2.3.9 limiting enrichment assays in the Seal Exhaust Station system and in the Evacuation Booster Station, respectively. TSR 2.2.3.10 deals with communication during heavy equipment handling (i.e., movement of large equipment over cells in the (1) Operating and (2) Floating operational modes. These TSRs are acceptable.

TSRs 2.2.3.4 and 2.7.3.3 require the fire suppression system to be operable in all operational modes except in the Shutdown mode while the compressor motor lube oil is valved off or removed. Appropriate compensatory actions, which include performing hot work only when adequate portable fire suppression equipment is available, providing for fire patrols, and system restoration requirements are included for the period of inoperability. Appropriate surveillance requirements are established.

Several TSRs have been established to prevent explosions and highly exothermic reactions in the cascades that could result in UF_6 releases and the spread of contamination. TSRs 2.2.3.5 and 2.7.3.4 require the removal of coolant from cells and freezer/sublimers before any addition of oxidants to prevent the possibility of mixing coolant (R-114) with treatment gasses (oxidants such as ClF_3 and F_2) in sufficient quantities that could result in an explosion in the presence of an ignition source. TSR 2.7.3.7 requires addition of oxidants to be administratively controlled to prevent the possibility of attaining explosive mixtures in the purge cascade. TSRs 2.2.3.6 and 2.7.3.5 require cell treatment to be monitored with an Infrared Analyzer to ensure that (1) adequate amounts of ClF_3 are present to maintain low concentrations of its highly exothermic reaction products, and (2) no reactive quantities of hydrocarbons are present during cell treatment. TSRs 2.2.3.7 and 2.7.3.6 require cells to be isolated prior to introducing treatment gasses when using the inverse recycle cell treatment method. TSR 2.2.3.11 limits the concentration of ClF_3 and/or F_2 to less than 8 mole % in material pumped from the Evacuation Booster Station (EBS). This TSR prevents ClF_3/F_2 and R-114 from reacting explosively in the EBS cooler in the event of a coolant (R-114) leak and presence of an ignition source. TSR 2.7.3.8 prevents the operation of two Freon Degraders at one time to keep concentrations of F_2 from exceeding 16 mole % in the purge cascade. TSR 2.7.3.9 limits the addition of F_2 to the Freon

Degrader to 400 scfd. This rate ensures that oxidant concentration does not exceed the minimum oxidant concentration of 19 mole % that could result in a highly exothermic reaction in the presence of an ignition source in the purge cascade operating at a high-side pressure of 14.45 psia (cascade pressure LCO). Appropriate LCOs, potentially abnormal conditions, required actions, and surveillance requirements have been established for the TSRs to prevent explosions and highly exothermic reactions in cascade equipment.

Removal of equipment with deposits greater than safe mass is covered by TSRs 2.2.3.16 and 2.7.3.15. These TSRs require a presurvey and post survey of equipment to be removed, to determine if uranium deposits exist which would require special handling, openings to be covered or closed, and decontamination within 72 hours. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established. TSRs 2.1.3.18, 2.2.3.12, 2.5.3.14, and 2.7.3.11 appropriately prohibit the use of direct heat on a solidified UF_6 plug until flow clarity has been assured.

Section 2.3 contains the TSRs for the freezer/sublimers (F/S) in the cascades. USEC has established appropriate operational modes in TSR 2.3.1. An appropriate safety limit of 11,900 pounds UF_6 has been established (2.3.2.1). The F/S weight trip system (2.3.3.1) prevents overfilling the vessel with UF_6 by isolating the F/S when the amount of UF_6 in the vessel exceeds the LCS value of 9,000 $\pm$456 pounds. TSR 2.3.3.3 requires the vent path from the F/S to the cascade to be open to prevent rupturing of the F/S vessel and a subsequent UF_6 release. Overpressurization of the UF_6 system may be caused by the flashing of R-114 inside the F/S following failure of the high pressure R-114 coolant tubes. In TSR 2.3.3.2, USEC has established an appropriate assay limit applicable during the UF_6 solidification process. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established.

Section 2.4 contains the TSRs for cold trapping of UF_6 in the cascades. USEC has established appropriate operational modes in TSR 2.4.1. TSR 2.4.3.2 requires the pressure relief system of the cold trap to be operable in the Flash (UF_6 vaporization) mode to prevent rupturing of the trap shell with a subsequent UF_6 release due to overpressurization caused by (1) incorrect valving and other operator errors during the flashing operation or by (2) liquefying or solidifying a gas other than UF_6. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established. USEC has established appropriate assay limits in TSRs 2.4.3.3 and 2.4.3.4, for cold recovery freezing operations in X-330 and X-333, and for venting through Alumina Traps in X-333. Appropriate surveillance requirements have been established. TSR 2.4.3.5 has been established to prevent an explosion in the cold trap and a subsequent release of UF_6 and treatment gasses. This is accomplished by ensuring that prior to entering the Freeze operational mode (1) the coolant (R-114) concentration is limited to 16.9 mole %, (2) the ClF_3/F_2 concentration is maintained between 0.14 and 5.25 mole %, (3) the cold trap inlet pressure is limited to a maximum of 3.3 psia and (4) the cold trap shell temperature is maintained above -65 °F. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established.

TSR 2.4.3.1a requires the detection capabilities of the criticality accident alarm systems to be operable for the cold recovery areas in X-330 and X-333 Buildings, when equipment or processes contain greater than 700 grams of ^{235}U at an enrichment greater than or equal to

1.0 weight % ^{235}U. TSR 2.4.3.1.b requires the alarming capabilities of the criticality accident alarm systems to be operable in areas where the maximum foreseeable absorbed dose in free air would exceed 12 rad. Appropriate surveillance requirements are established.

The TSRs for the product and tails withdrawal operations are contained in TSR 2.5. USEC has established appropriate operational modes (TSR 2.5.1) for the activities that occur in the ERP, Low Assay Withdrawal (LAW) and Tails Withdrawal (Tails) facilities. Safety limits for the UF_6 condenser coolant pressures of 330 psig for LAW and 440 psig for ERP (TSR 2.5.2.1) have been established at 110% of the MAWP to ensure the integrity of the UF_6 condensers. USEC has not proposed an SL for Tails because the UF_6 pressure boundary would not be threatened if coolant enters the UF_6 side. This is because unlike R-114 coolant which is used in ERP and LAW, the coolant used at Tails has a low enough vapor pressure to maintain it as a liquid when forced into the UF_6 system. TSR 2.5.3.2 requires the R-114 coolant overpressure system to be operable in the (1) Withdrawal, (2) Compression/Liquefaction and (3) Standby operational modes, establishes LCS pressure values at 100% of the MAWP (400 psig for ERP and 300 psig for LAW) and sets the rupture disk actuation pressures at the LCS values with a 5% burst tolerance. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been established.

TSR 2.5.3.1a requires the detection capabilities of the criticality accident alarm systems to be operable in the ERP, LAW and Tails Stations, when equipment or processes contain greater than 700 grams of ^{235}U at an enrichment greater than or equal to 1.0 weight % ^{235}U. TSR 2.5.3.1.b requires the alarming capabilities of the criticality accident alarm systems to be operable in areas where the maximum foreseeable absorbed dose in free air due to criticality would exceed 12 rad. Loss of detection or alarming capabilities requires immediate discontinuation of movement of cylinders with UF_6, and waste containing uranium enriched to 1.0 weight % ^{235}U or more. However, the current or ongoing withdrawal cycle is permitted to be completed to avoid a competing risk that may be introduced by terminating withdrawal operations. In addition, only personnel equipped with alternate means of alarm notification are allowed in the area for which coverage has been lost. Restoration of criticality detection and alarming capabilities is required within 48 hours. Appropriate potentially abnormal conditions, required actions, and surveillance requirements are established.

TSRs have been established for the operability of the UF_6 release detection systems and associated isolation systems. TSR 2.5.3.3 has been established for the UF_6 (Smoke) detectors located in ERP, LAW and Tails withdrawal areas inside and outside compressor housings and in the condenser/accumulator areas. These detectors, upon detection of a UF_6 release, will alert the operator by sounding an alarm in the ACR. TSR 2.5.3.4 has been established for smoke detectors associated with the pigtail line isolation system. These detectors located over each withdrawal position will automatically isolate the withdrawal position (pigtail) upon detection of UF_6 to limit the release quantity. Both TSRs have appropriate potentially abnormal conditions, required actions, and surveillance requirements.

USEC has established TSR 2.5.3.5 for monitoring the assay of uranium above 1.0 weight % during UF_6 compression/liquefaction and withdrawal operations to ensure that the maximum allowable enrichments are not exceeded. An appropriate LCO, potentially abnormal conditions, required actions, and surveillance requirements have been established. TSR 2.5.3.6 establishes appropriate assay limits for the accumulators at ERP, LAW and Tails. TSR 2.5.3.7 establishes appropriate assay limits for 2.5, 10 and 14 ton cylinders and condenser pressure limit (to ensure any HF present is maintained in vapor form) at ERP, LAW and Tails. Appropriate potentially abnormal conditions, required actions, and surveillance requirements have been set.

Cylinder filling and handling TSRs have been established for several of the facilities. TSR 2.5.3.8 prohibits the cylinder scale cart movement in the Withdrawal operational mode while the cylinder is attached to the withdrawal manifold. Prior to removal of a cylinder from a scale cart, the LCO in TSR 2.5.3.11 requires 2.5, 10 and 14 ton UF_6 cylinder fill weights to be less than or equal to the listed cylinder standard fill weights. This TSR also includes appropriate functional testing surveillance requirements for the scales. Normal heating of a cylinder that contains more than the standard fill limit is prohibited by TSR 2.1.3.8. TSR 2.1.3.17 limits the amount of unknown material in a cylinder to be filled by prohibiting cylinder heating if an unexplained cylinder weight discrepancy of greater than 40 pounds exists between shipper UF_6 weight and received cylinder weight. Cylinder handling TSRs require cylinders containing liquid UF_6 to be moved only by overhead cranes that are operable (2.1.3.12 and 2.5.3.10), scale carts and railcars (2.1.3.14 and 2.5.3.9) and limit movement of one cylinder over another containing liquid UF_6 (2.1.3.16 and 2.5.3.12). Appropriate potentially abnormal conditions, required actions and surveillance requirements have been established. All of these TSRs are acceptable.

Section 2.6 contains TSRs for the decontamination facility (X-705). USEC has established appropriate operational modes in TSR 2.6.1. Appropriate safety limits have been established for the calciner internal temperature (2.6.2.1), microfiltration effluent pH (2.6.2.2) and the microfiltration effluent bag filter differential pressure (2.6.2.3).

TSRs 2.6.3.4a and 2.8.3.1a require the detection capabilities of the criticality accident alarm systems to be operable in X-705 and X-710 (laboratory) Buildings, when equipment or processes contain greater than 700 grams of ^{235}U at an enrichment greater than or equal to 1.0 weight % ^{235}U. TSRs 2.6.3.4.b and 2.8.3.1b require the alarming capabilities of the criticality accident alarm systems to be operable in areas where the maximum foreseeable absorbed dose in free air due to criticality would exceed 12 rad. Loss of detection or alarming capabilities requires immediate discontinuation of operations with fissionable material. In addition, only personnel equipped with alternate means of criticality alarm notification are allowed in the area for which coverage has been lost. Restoration of criticality detection and alarming capabilities is required prior to reinitiating activities required to be monitored for criticality. Appropriate potentially abnormal conditions, required actions, and surveillance requirements are established.

In addition to the administrative TSR 3.11, specific TSRs for preventing criticality have been established. TSR 2.6.3.1 requires the calciner high temperature shutoff system to be operable in the (1) Operating and (2) Standby (heaters energized) operational mode. This system closes the inlet feed valve and de-energizes the heaters when the temperature

reaches the LCS of 1800°F thus ensuring that the SL of 2000°F is not exceeded. Calciner tube burn-through could result in the accumulation of an unsafe geometry of uranium and in spread of contamination. TSR 2.6.3.2 requires the calciner discharge collector probe detection system to be operable in the (1) Operating and (2) Standby operational modes. This system ensures that the uranium oxide product does not exceed safe geometry in the calciner discharge throat by closing the inlet feed valve and sounding an alarm when the uranium oxide contacts the probe. TSR 2.6.3.3 requires the calciner can level probe detection system to be operable in the (1) Operating and (2) Standby operational modes except when the discharge valve is closed. This system ensures that the uranium oxide does not overflow out of the can collar into an unsafe geometry by closing the inlet feed valve and sounding an alarm when the uranium oxide contacts the probe. TSR 2.6.3.7 requires the calciner tube rotation interlock system to be operable while material is being processed through the system. The interlock ensures that the calciner tube does not fail from being heated unevenly, by closing the inlet feed valve and de-energizing the heaters when tube rotation stops thus preventing concentrated uranyl nitrate and uranium oxide from potentially leaking into an unsafe geometry. Appropriate LCOs, abnormal conditions, required actions and surveillance requirements have been established for the TSRs associated with the calciner.

TSRs for preventing criticality have been established for microfiltration operations. TSR 2.6.3.5 requires the microfiltration pH shutoff system to be operable while material is being processed through the system. This system ensures that the pH remains above the SL of 6.5 by stopping flow of solution to the effluent tank (unfavorable geometry) when the pH drops below the LCS value of 7.0. TSR 2.6.3.6 requires the microfiltration permeate effluent bag filter to be operable while material is being processed through the system. This system closes the inlet valve to the effluent tank when the differential pressure across the bag filter reaches 17 psid (LCS) thus ensuring that the SL of 30 psid is not exceeded. At these differential pressures, an insufficient quantity of uranium is contained in the bag filters to result in a criticality in the effluent tank upon their accidental rupture. Appropriate LCOs, abnormal conditions, required actions and surveillance requirements have been established for the TSRs associated with the microfiltration system.

TSRs are also established to limit ^{235}U quantities to no greater than 350 grams in batch operations and unfavorable geometry containers associated with the Oil and Grease Removal Unit (2.6.3.8) and the Microfiltration Effluent Filter Press (2.6.3.9). Appropriate LCOs, abnormal conditions, required actions and surveillance requirements have been established for these TSRs.

USEC has also established minimum design requirements (Design Features) for UF_6 cylinder lifting fixtures (2.1.4.1, and 2.5.4.1), UF_6 cylinders (2.1.4.2 and 2.5.4.2), cylinder pigtails (2.1.4.3 and 2.5.4.3), scale pit Raschig Rings (2.5.4.4), overhead cranes (2.1.4.6 and 2.5.4.5), X-342 condensate sump and oil interceptor Raschig Rings (2.1.4.4), autoclave shell (2.1.4.5), seal exhaust pump oil overflows (2.2.4.1 and 2.7.4.1), handtable overflows (2.6.4.1), air gaps (2.6.4.2), dike height (2.6.4.3), and tank covers (2.6.4.4). All of the design features and surveillance requirements are appropriate.

The staff has reviewed the TSRs for PORTS and concludes that based upon the safety basis provided by USEC as part of the certification application, they establish the

necessary controls, provide the necessary program commitments for the facility, and meet the requirements of 10 CFR Part 76 and are, therefore, acceptable. The SAR upgrade project may result in identification of new safety systems and the need for additonal TSRs; it could also result in the elimination of current safety systems and a recommendation to eliminate some of the TSRs. Once the TSR upgrade project is complete, the TSRs will need to be reevaluated based on any new information.

While USEC will be allowed to make changes to the TSR basis statements in accordance with the plant change control process described in Section 6.3 of the SAR, USEC should not make changes to the TSRs themselves without prior staff approval. Accordingly the staff recommends the following certificate condition:

> USEC shall conduct its operations in accordance with the Technical Safety Requirements that are contained in Volume 4, Revision 5 of the Application dated August 1, 1996, as modified by Revision 6 of the Application dated August 12, 1996. Changes to the Technical Safety Requirements shall require NRC approval prior to implementation.

Chapter 7 RADIATION PROTECTION

The regulations at 10 CFR §76.60(d) require USEC to comply with the applicable provisions of 10 CFR Part 20. In accordance with 10 CFR §20.1101, USEC is required to develop, document, and implement a radiation protection program to ensure compliance with the provisions of 10 CFR Part 20. The PORTS radiation protection program is described in SAR §5.3. TSR 3.13 commits USEC to establish, implement, and maintain the program described in the SAR. The TSR also requires the following elements to be addressed in the program: HP technicians training and qualifications, personnel exposure control and measurement, contamination control, radioactive material control, radiological protection instruments and equipment, and records and reports.

The objective of the radiation protection program is to provide adequate protection of the PORTS work force under normal conditions of operation and following accidents. The following sections discuss the types of radioactivity and radiation that will be encountered at the site, the radiation protection program for workers, and the NRC staff's evaluation of the program.

7.1 Radiation and Radioactivity Sources

The predominant radioactive material utilized at the site is natural, low-enriched, and depleted uranium primarily in the form of uranium hexafluoride (UF_6). However, other uranium compounds (for example, in wastes) are also present as gases, liquids, and solids. Natural uranium is about 99.3% uranium-238 (U-238), and about 0.71% uranium-235 (^{235}U).

In depleted uranium tails, the ^{235}U content is reduced to about 0.2 to 0.47 weight %. The numerous daughters, such as thorium and radium radionuclides, found accompanying natural uranium are removed in the uranium milling process prior to being sent to the PORTS. Incidental radioisotopes that accompany uranium are shorter-lived daughters (for example, thorium-234 [Th-234] and metastable protactinium-234 [Pa-234m]) which would "grow" into partial or complete radioactive equilibrium with their long-lived parents.

Most of the radiation emitted by the radioisotopes encountered are alpha and beta particles which are non-penetrating forms of radiation that are shielded from workers by UF_6 cylinder shells and primary containment systems (for example, process lines). Due to the high density of UF_6 when stored as a solid, the material also provides considerable self-attenuation of x-rays and gamma rays from the uranium series nuclides present. However, large, newly emptied UF_6 cylinders are devoid of this self-attenuation. Consequently, significant shielding of the relatively high gamma energy photons emanating from the short-lived U-238 daughter products is not provided. This results in some of the highest direct radiation dose rates at the plant (100 millirem per hour for a short period of time at the bottom of a newly emptied 14-ton cylinder). A significant portion of the direct radiation encountered at the PORTS is in the form of bremsstrahlung radiation, which is generated by the interaction of beta radiation with high atomic number atoms, such as uranium in UF_6 and, to a lesser extent, iron in cylinders. Full UF_6 shipping cylinders exhibit an external dose rate of about 5 millirem per hour on contact (NRC, 1994). In addition to alpha decay, U-238 also spontaneously fissions with a yield of 0.0054%. Therefore, large

quantities of uranium can also result in small neutron dose rates. For instance, in the tails cylinder storage areas, USEC reports a neutron dose rate of between 0.1 and 0.3 mrem/hr at 30 cm.

Similar to the low external radiological dose hazard at the PORTS, the internal radiological dose hazard is also low. The emphasis applied to containment of UF_6 and application of the radiation protection (RP) program to meet uranium's toxicity limit in 10 CFR Part 20 at the PORTS, ensures that internal radiological exposures are minimized.

A primary concern for most the PORTS operations is incidental or accidental inhalation of uranium, which can, if not treated, cause non-stochastic chemical damage to the kidney (nephrotoxicity) if intakes exceed a threshold within a specified period of time. Significant releases of UF_6 to most work areas are unlikely, since a large part of the cascade is operated below a pressure of 1 atmosphere so that leaks are into the system, and not from the system into the process areas. In addition, the process areas are large and routinely unoccupied. Also a UF_6 release detection system exists with detector heads located and required to be operable in those portions of the cascade that operate above atmospheric pressure. Most of the other sources of radioactivity utilized at the facility are small calibration and radiochemistry (quality control) standards which when used correctly pose a small radiation exposure risk to workers and none to the public. Byproduct material may be in solid, liquid, or gaseous form, and is not necessarily restricted to sealed sources.

7.2 Radiation Protection Program

The PORTS radiation protection program involves the entire range of facility operations which could affect worker safety pertaining to radioactive material in normal operations or during accident conditions. Program elements include: (1) Policy statements; (2) Procedures implementing the program; (3) Training including General Employee Radiological Training (GERT); (4) Radiological surveys; (5) Access control; (6) Administrative dose levels; (7) Adherence to procedures and radiological controls; (8) Use of protective clothing and personal protective equipment; (9) Audits and inspections; (10) Personnel exposure monitoring; (11) Recordkeeping; and (12) Reports.

The RP Manager is required to annually review the radiation protection program and generate a report which is reviewed by the ALARA Committee - a subcommittee of the PORC. In addition, the radiation protection program is also audited as described in Section 3.7 of this CER.

7.2.1 ALARA

NRC regulations require that occupational exposures be as low as is reasonably achievable (ALARA). USEC is committed to maintaining radiation exposures in accordance with the ALARA principle. The responsibility for establishing the ALARA policy rests with the Executive Vice President, Operations. The General Manager has the overall responsibility and authority for the ALARA program and the RP Manager (or designee) is responsible for implementing the ALARA program. The staff finds USEC's proposed ALARA policy to be

in accordance with 10 CFR Part 20 and its associated responsible functional positions to be adequate.

USEC is committed to establishing an ALARA Committee. The ALARA committee's authority is limited to reviews and recommendations only. Included in the Committee's charter for the purpose of maintaining occupational doses ALARA are; (1) annually evaluating the implementation of the ALARA program; (2) establishing the annual radiation exposure goals and monitoring radiation exposure and airborne activity trends; (3) advising the PORTS management and the PORC for the purposes of maintaining occupational doses ALARA; (4) reviewing proposed design changes with a projected collective dose greater than 1 person-rem; (5) reviewing practices resulting in a collective dose greater than 1 person-rem; and (6) reviewing revisions to ALARA procedures. Being a subcommittee of the Plant Operational Review Committee (PORC), the ALARA Committee chair (RP Manager) reports to the PORC chair.

Membership of the ALARA Committee includes representatives from Environmental, Safety, and Health, Cascade Operations, Chemical Operations, Feed and Transfer Operations, Production Support, Maintenance, Nuclear Regulatory Affairs, Training and Procedures, Engineering, Site and Facilities Support, United Plant Guard Workers of America Bargaining Unit, and Oil, Chemical, Atomic Workers Bargaining Unit. A quorum consists of the chair or vice-chair (Production Support Manager) and five members. The ALARA committee meets at least semiannually and as directed by the chair. Meeting minutes are provided to each ALARA Committee member and the PORC. The NRC staff finds the ALARA Committee's makeup and charter adequate to appropriately implement the ALARA principle at PORTS.

7.2.2 Responsibilities

The RP Manager directs the RP organization and provides technical oversight of all radiological protection procedures. The RP Manager, as well as the HP support personnel (HP technicians and their supervisors), have the authority to stop work to maintain the integrity of the RP program. The RP Manager has direct access to the General Manager and the Enrichment Plant Manager. Some of the RP Manager's duties stated in Section 6.1.1 of the PORTS SAR include training of personnel in the use of radiological program support equipment, determining the need for issuing and closing out radiation work permits, and conducting the radiological occupational monitoring program. Some of the RP support personnel duties include dosimetry, bioassay, instrumentation and calibration functions, and the guiding of workers in the radiological aspects of the job.

The minimum qualifications of the RP Manager or his/her designee are a Bachelors degree in engineering, health physics, radiation detection, or the physical sciences or equivalent technical experience, and four years experience in radiation protection including 6 months at a uranium processing facility. Section 5.3.1.4 of the PORTS SAR states that HP technicians and their supervisors are required to have technical qualifications pertinent to their assigned duties; training includes initial, on-the-job and continuing training. Table 5.3-11 of the PORTS SAR provides the HP technician course curriculum. The staff finds the qualification requirements for the RP organization which includes the RP Manager, and the

HP technicians and their supervisors adequate and consistent with the requirements of 10 CFR Part 20 and 10 CFR Part 76.

7.3 Occupational Radiation Protection

USEC has defined in the PORTS SAR the following radiological areas for protection of workers from the chemical toxicity of uranium and from radiation:

Radioactive Material Areas (RMAs): Includes areas or rooms containing radioactive material more than 10 times Appendix C to 10 CFR §§20.1001-20.2401. For U-238, this threshold corresponds to 1000 μCi or 30 grams, for U-234 this threshold corresponds to 0.01 μCi and for natural uranium this threshold corresponds to 55 mg.

Contamination Control Zones (CCZs): Includes areas generally with removable surface contamination levels up to those provided in PORTS SAR Table 5.3-2. Certain discrete areas may exceed these levels. For uranium, the level would be 1,000 dpm/100 cm^2.

Contamination Areas (CAs): Includes areas with removable surface contamination levels between 1 and 100 times those provided in PORTS SAR Table 5.3-2 averaged over 1 m^2. For uranium this range is between 1,000 and 100,000 dpm/100 cm^2 averaged over 1 m^2.

High Contamination Areas (HCAs): Includes areas with removable surface contamination levels above 100 times those provided in PORTS SAR Table 5.3-2. For uranium the levels would be above 100,000 dpm/100 cm^2 averaged over 1 m^2.

Fixed Contamination Areas: Includes areas with removable surface contamination levels less than those provided in PORTS SAR Table 5.3-2 but total (removable + fixed) surface contamination levels above those provided in PORTS SAR Table 5.3-2. For uranium, the contamination level corresponds to 1,000 dpm/100 cm^2. Fixed Contamination Areas also include areas with direct radiation levels greater than 50 μrem/hr at 1 meter from the surface.

Soil Contamination Areas: Includes areas with volumetric uranium concentration levels greater than 30 pCi/g or Tc-99 concentrations greater than 70 Pci/g.

Airborne Radioactivity Areas (ARAs): Includes areas with potential to exceed airborne soluble uranium concentrations levels greater than 50 $\mu g/m^3$.

Radiation Areas (RAs): Includes areas with direct radiation levels between 5 and 100 mrem/hr at 30 cm from the surface.

High Radiation Areas (HRAs): Includes areas with direct radiation levels greater than 100 mrem/hr at 30 cm from the surface.

Very High Radiation Areas: Includes areas with direct radiation levels greater than 500 rads/hr at 1 m from the surface.

USEC will post caution signs for RMAs, ARAs, Ras, and HRAs in accordance with the posting requirements of 10 CFR Part 20.

The staff has reviewed access, posting and monitoring requirements for Ras provided in PORTS SAR Section 5.3.3 and has determined them to be adequate and consistent with the requirements of 10 CFR Part 20.

7.3.1 Exposure Controls and Exposure Experience

The 10 CFR Part 20 annual limits for workers which USEC will be required to comply with are summarized below:

EXPOSURE CONDITIONS	PART 20 RADIOLOGICAL EXPOSURE LIMITS
External Radiation	5 rem/yr total effective dose equivalent (TEDE); includes summation of both external deep dose equivalent and internal committed effective dose equivalent (CEDE). Internal dose equivalents for each organ are multiplied by risk-based weighting factors and summed (except for lens of eye, skin and extremities).
	Lens of eye: 15 rem/yr
	Hand, elbow, arm below elbow, foot, knee, and leg below knee: 50 rem/yr shallow dose equivalent. Same limit for skin, with requirement for calculating maximum skin dose to 1 cm^2 area.
Internal Radioactivity	Annual limit of intake (ALI) based on exposure to 2,000 derived air concentration (DAC) - hours per year.
	Organs are assigned weighting factors based on estimated risk/rem to that organ versus risk/rem for whole body irradiation, capping the dose limit at 50 rem/yr to avoid non-stochastic effects. The inferred limits are: Gonads: 20 rem/yr Breast: 33 rem/yr Red Marrow: 42 rem/yr Lung: 42 rem/yr Thyroid: 50 rem/yr Bone Surface: 50 rem/yr Each of 5 highest remaining Organs: 50 rem/yr Embryo/Fetus: 0.5 rem TEDE/yr

To protect workers from chemical toxicity effects from inhalation of soluble (Class D) uranium, 10 CFR §20.1201(e) also limits worker intake to no more than 10 mg of soluble uranium in a week.

In addition to meeting the NRC 10 CFR Part 20 requirements, USEC has established an annual administrative radiation dose limit of 500 millirem TEDE. This is acceptable to the NRC staff.

External radiation monitoring dosimeters (NVLAP-accredited thermoluminescent dosimeters, or TLDs) are issued to monitor external exposures and exchanged at least quarterly (±2 weeks). Other dosimeters such as finger rings, neutron dosimeters and direct-reading dosimeters are issued where the standard TLDs cannot provide the desired information or are not practical.

USEC states in Section 5.3.2.3 of the PORTS SAR that internal exposures for workers will be evaluated by bioassay procedures to determine intakes based on the biokinetic model in NUREG/CR-5566 "Evaluation of Health Effects in Sequoyah Fuels Corporation Workers from Accidental Exposure to Uranium Hexafluoride" and that air sampling will only be used to trigger special bioassay sampling. USEC is committed to performing bioassay for all personnel whose routine duties require entry into CCZs, CAs, HCAs, or ARAs and who have a potential to exceed doses of 100 millirem per year or 1 mg soluble uranium per week. The staff finds this criteria for performing bioassay and the action levels proposed in PORTS SAR Table 5.3-4 adequate to ensure compliance with the 10 CFR Part 20 dose limits.

Since the primary internal radiological dose hazard at PORTS is uranium in soluble form (Class D) and enrichment is generally less than 5% based on nuclear power reactor fuel requirements, meeting the 10 mg weekly intake limit over the entire year provides reasonable assurance that the annual limit on intake (ALI) will not be exceeded. To put this in perspective, inhalation of 10 mg of depleted uranium with an activity median aerodynamic diameter (AMAD) of 1.0 μm would result in a CEDE of less than 10 millirem. At an enrichment of 10% ^{235}U, the radiological dose would be about 120 millirem. AT an

USEC proposes to routinely collect monthly or quarterly bioassay (urine) samples for routinely determining uranium intakes. The lowest action levels proposed by USEC for monthly and quarterly samples are 5 and 0.5 μg U/l, respectively. Using ICRP Report 54, "Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation," which utilizes the current International Commission on Radiological Protection (ICRP) metabolic models, the NRC staff estimates that a concentration of 5 μg U/l monthly lower limit of detection (LLD) in a 24-hour urine sample (1.4 liter) collected during the tenth, twentieth, and thirtieth (bounding) day following a single intake (conservative), corresponds to intakes of about a 1, 2 and 4 mg of soluble uranium, respectively. Similarly, a concentration of 0.5 μg U/l (quarterly LLD) in a sample collected after ninety (bounding) days, corresponds to a single (conservative) intake of about 5 mg of soluble uranium. These are well below the 10 mg weekly intake limit. It should be noted that as discussed in Chapter 5 of this CER, the low thresholds of sight and smell associated with UF_6 releases would cause an exposure level, well below those discussed above, to be

known requiring a special bioassay evaluation. Therefore, not only is it highly conservative to assume a single intake occurring on the first day of the bioassay sampling cycle, but it is also highly conservative to assume that the single intake would go undetected.

Based on the metabolic model for uranium presented in ICRP-26 (ICRP, 1977) and ICRP-30 (ICRP, 1979), which provides data for an AMAD of 1.0 μm, the NRC staff calculated an approximate 30% increase in uranium deposition in the kidney for a 0.2 μm AMAD particle size. During normal PORTS operations, the NRC staff expects any uncontainment of uranium to result in particulates with AMADs greater than 0.3 μm. In addition, an evaluation of 31 workers accidentally exposed to natural uranium in 1986 indicates that the ICRP guidance may overestimate the amount of uranium present in urine at 7 days, since bioassay data indicates more rapid excretion of Class D uranium than originally believed (Fisher et al, 1990). This implies that the potential intake which might go undetected under the proposed bioassay program could be somewhat higher than that estimated using the ICRP model. Nevertheless, the NRC staff concludes that it is unlikely that these potential nonconservatisms would practically account for a large enough factor so as to result in PORTS not being able to detect weekly intakes involving more than 10 mg of soluble uranium.

As stated by USEC in Table 5.3-6 of the PORTS SAR (USEC, 1996), during the years 1992, 1993 and 1994, there were five individual external exposures greater than 500 millirem and the largest internal radiological dose for those years was 79 millirem CEDE. It should be noted that 10 CFR Part 20 requires radiation monitoring if there exists a potential for exceeding 10% of the annual occupational dose limits which includes a 5,000 millirem TEDE limit. Collective radiological doses for 1992, 1993 and 1994 were 22, 34 and 31 person-rem, respectively.

7.3.2 Airborne Radioactivity in the Workplace and Ventilation Systems

USEC uses fixed and portable air-sampling equipment and lapel samplers in accessible areas where the airborne concentrations could exceed 10% of the DAC (PORTS SAR Table 5.3-5) averaged over 8 hours. This is acceptable to the NRC staff. Continuous air samplers (CAMs) are located throughout the PORTS in Buildings X-326, X-330, X-333, X-342, X-343, X-344, X-345, X-705, X-710 and X-744G. Since bioassay and not air sampling is the primary method for determining internal dose, air sampling need not be representative. Nevertheless, the RP Manager reviews the appropriateness of the air sampling locations annually. Fixed air sample media are changed daily during the work week. However, sample filters may be exchanged sooner if needed. This is also acceptable to the NRC staff.

USEC uses portable high efficiency particulate (HEPA) filter systems for short duration jobs when an unprotected worker may potentially exceed a 0.8 DAC-hr exposure. This criterion of 0.8 DAC-hr is acceptable to the NRC staff, since it corresponds to an intake of 1.3 mg or less of soluble uranium. The differential pressure for portable HEPA filter systems used for radiological protection is checked during every operating procedure. The operating pressure range is required to be in accord with the manufacturer's recommendations or as specified in the technical design basis. The NRC staff finds this acceptable.

HEPA filter systems are dioctyl phthalate (DOP) or equivalently tested every eighteen months. This frequency is in accordance with ASME N510-1989 entitled "Testing of Nuclear Air Treatment Systems." This is acceptable to the NRC staff.

USEC is committed to maintaining the average air velocity above 100 fpm through openings of hoods used to sample uranium or containing readily dispersible uranium. The NRC staff finds this acceptable and consistent with good industrial safety practice.

USEC is committed to maintaining glove boxes that could generate airborne radioactivity at a negative differential pressure of at least 0.25 inches of water while the glove box is in use. The NRC staff finds this acceptable and consistent with good industrial safety practice.

Ventilation equipment is typically designed such that normal air flow or leakage flows are generally from areas of lesser contamination to areas of higher potential contamination. However, in the PORTS process buildings, general air flow is from the cell floor which has a comparatively higher potential for contamination and UF_6 releases to the ground (operating) floor. However, the combination of (1) use of continuous air samplers, (2) UF_6 release detectors in heated enclosures being able to detect UF_6 releases as small as a few pounds, (3) periodic contamination surveys, (4) low numbers of workers present in the process buildings, (5) most of the cascade operating below 1 atmosphere, (6) worker protection requirements as part of the TSRs, and (7) large mixing volumes of the cell and ground floors, contribute to lessen the hazard of this unfavorable ventilation scheme. Considering all of these factors, and operational history of PORTS, the NRC staff does not find this unfavorable design of the process building ventilation system to result in a significant hazard to workers present in the Operating Floor.

7.3.3 Control of Surface and Personnel Contamination

In areas restricted for the purposes of radiological control, USEC proposes that worker exposure to surface contamination is or will be minimized by proper use of surveys, posting, protective clothing and equipment. USEC is committed to limiting skin or personal clothing contamination at egress from radiological areas to no more than 1,000 dpm/100 cm^2 alpha, and 5,000 dpm/100 cm^2 beta/gamma. USEC's action levels for surface contamination on laundered protective clothing are 1,000 dpm/100 cm^2 alpha, and 20,000 dpm/100 cm^2 beta/gamma. These action levels are acceptable to the NRC staff. It should be noted that most of the radioactive material that could contaminate protective clothing at PORTS will be in soluble form and therefore will be readily removed by laundering.

USEC is committed to providing routine contamination survey monitoring in Lunchrooms/Breakrooms located within Restricted Areas on a daily basis; permanent boundary control stations on a weekly basis; Feed/Withdrawal Stations, change rooms, and UF_6 sample handling laboratories on a monthly basis; CCZs and contaminated maintenance areas on a quarterly basis; and the remaining uranium processing areas on an annual basis. Based on the results of the surveys, Radiological Areas will be reposted or decontaminated in accordance with the ALARA principle, as necessary. Even though the survey frequencies proposed by USEC are generally lower than those recommended in Regulatory

Guide 8.24 entitled "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication" dated October 1979, the staff finds it acceptable for the following reasons: strict requirements related to the containment of UF6, use of continuous air samplers throughout PORTS, UF_6 release detectors in uranium processing areas being able to detect UF_6 releases as small as a few pounds, low numbers of workers present in the process buildings, most of the cascade operating below 1 atmosphere, and large mixing volumes in most of the uranium processing areas.

USEC is required by regulation to survey incoming and outgoing shipments of radioactive material. In addition, USEC is committed to restricting release of materials and equipment for unrestricted use if removable surface contamination levels exceed those presented in PORTS SAR Table 5.3-2. The NRC staff finds this acceptable and in accord with NRC guidance related to unrestricted release of equipment and material.

The NRC staff finds USEC's program to control surface and personnel contamination acceptable. The basis for accepting USEC's proposed surface contamination criteria is as follows. Most of the contamination present at PORTS is in the form of natural, enriched and depleted uranium. For removable surface contamination (U-238) of 1,000 dpm/100 cm^2 averaged over an entire facility, using a resuspension factor of $5x10^{-5}$ per meter (IAEA, 1970), the NRC staff calculated a weekly intake (40-hour exposure) via inhalation of less than 0.4 mg of uranium. For fixed uranium surface contamination of 5,000 dpm/100 cm^2 (5% enrichment), and assuming an infinite planar source and 100% occupancy, the NRC staff calculated an annual deep dose equivalent of less than 1 millirem.

7.3.4 Respiratory Protection Program

USEC utilizes respiratory protection at PORTS when engineering and administrative controls cannot prevent the potential for airborne contamination. 10 CFR Part 20, Subpart H, provides the requirements for an acceptable respiratory protection program. As stated, in 10 CFR Part 20 and PORTS SAR Section 5.3, respiratory protection is only to be relied upon when process or other engineering controls are impracticable. The respiratory protection program at PORTS is administered by the Industrial Hygiene (IH) group.

Respirators are used (1) upon entering ARAs, (2) during breach of containment systems, (3) when removable surface contamination exceeds 100 times the values listed in PORTS SAR Table 5.3-2 (e.g. 100,000 dpm/100 cm^2 for uranium), and (4) when work on contaminated surfaces has the potential to generate airborne radioactivity. Use of respirators are considered when a worker can potentially be exposed to 0.8 DAC-hours during a work shift. For situations where certain physical limitations, such as heat stress, may contradict safety or not be in accordance with ALARA, the RP Manager could authorize non-use of respirators. These criteria and the condition for non-use of respiratory protection are acceptable to the NRC staff and consistent with the requirements of 10 CFR Part 20. The main reasons for this acceptability are as follows: (1) workers in ARAs without respiratory protection have the potential to intake more than 2.4 mg of soluble uranium; (3) for facilities with an average removable surface contamination (U-238) of 100,000 dpm/100 cm^2, application of a resuspension factor of $5x10^{-5}$ per meter (IAEA, 1970) would result in a weekly intake (40-hour exposure) via inhalation of less than 40 mg

of uranium; and (5) 0.8 DAC-hours during a work shift corresponds to an intake of less than 1.3 mg of soluble uranium.

7.3.5 Instrumentation, Calibration, and Maintenance Program

Monitoring instruments under the RP program at PORTS are used to (1) count air and smear samples, (2) measure direct dose rates (Mr/hr), (3) continuously monitor the air (alarming CAMs) and (4) detect skin and clothing contamination. Typical detection limits for these detectors (except for the direct dose rate meters) are presented in PORTS SAR Table 5.3-7.

10 CFR §20.1501(b) requires assurance that instruments and equipment used for quantitative radiation measurements (dose rate and effluent monitoring) are calibrated periodically for the radiation measured. The National Council on Radiation Protection and Measurement states that the required frequency of calibration ranges from once every few weeks to annually depending on the amount of use an instrument receives, the environmental condition it is used under, and the historical experience of each instrument type (NCRP, 1991).

NRC licensees must make routine survey measurements with reasonable accuracy and reliability. Reliability is a function of the detector systems, instrument usage, manufacturing quality, and the user's calibration and maintenance programs. USEC has proposed to calibrate instruments based on specifications derived from the applicable vendor manuals or other nationally recognized guidance (e.g. NCRP, 1991). In addition, USEC is committed to using calibration sources which are within ±5% of the stated value and have documented traceability links to National Institute of Standards and Technology with the exception of large area uranium slab sources, which are certified to ±10% of the stated value. USEC is also committed to calibrating air flow measurement devices for air samplers on an annual basis to within ±20% of the standard. The NRC staff finds the proposed instruments calibration requirements adequate and consistent with the requirements of 10 CFR Part 20.

The NRC staff has reviewed USEC's commitments related to the leak testing of sealed sources and found them to be acceptable and consistent with general NRC guidance.

USEC has proposed to perform source or operability checks of portable RP instruments on a daily basis or prior to use if not used on a daily basis. Operability checks of instruments, such as the hand and foot monitors, that have intrinsic functional test features are performed automatically on a continuous basis. In addition to continuous testing, these instruments are also source checked once a week. The NRC staff finds the proposed instrument operability checks to be adequate and consistent with the requirements of 10 CFR Part 20.

7.3.6 Radiation Work Permit System

USEC is committed to using Radiological Work Permits (RWPs) for work activities in Cas, HCAs, ARAs, Ras and HRAs. RWPs implement uranium's radiological and toxicity safety

requirements. The RP Manager designates RP personnel who are authorized to approve, issue, update, revise, and close RWPs. In PORTS SAR Section 5.3.1.6, USEC states that RP assesses the radiological conditions applicable to a task request. Since the same PORTS SAR section also states that RWPs are based on the toxicity of uranium, it may be inferred that RP would also assess the uranium toxicity conditions applicable to the task request.

General RWPs are issued for routine or repetitive activities. General RWPs are posted at access points or in centralized locations and are approved for periods not longer than 1 year.

Job-specific RWPs are issued for nonroutine operations or areas with changing radiological or uranium's toxicological conditions and remain in effect for the duration of the job or until closed by RP.

Based on its assessment of a job, RP includes information concerning protective clothing and special instructions such as radiological hold points, and approves and issues the RWP. RP closes the RWP upon work completion, RWP expiration or changes to the radiological or uranium's toxicological conditions of the job. Upon closure, RP evaluates the area to ensure it has returned to an acceptable condition. The NRC staff finds USEC's proposed system of RWPs adequate to protect workers from exceeding the radiological and uranium's toxicological limits contained in 10 CFR Part 20.

7.4 Exemptions from 10 CFR Part 20

USEC has requested an exemption from 10 CFR Part 20 requirements related to labeling containers. 10 CFR §20.1904 requires each container of radioactive material be labeled such that the radionuclide(s) including their estimated quantities, radiation levels, enrichment, and forms are identified. USEC states that it is impractical to label each container located in restricted areas. As a compensatory measure, USEC has proposed to place one caution sign in the area stating that every container may contain radioactive material. In addition, USEC is committed to surveying all containers removed from contaminated or potentially contaminated areas to ensure that contamination is not spread around the plant site. USEC has also requested an exemption from labeling UF_6 cylinders per 10 CFR §20.1904, since these are readily identifiable. As a compensatory measure, USEC has proposed to have UF_6 cylinders constantly attended by qualified Radiological Workers during movement. The staff finds the on-site radiological safety impacts that could result from this exemption to the requirements of 10 CFR §20.1904 to be minimal and recommends that this exemption be granted.

7.5 Items of Noncompliance

Compliance Plan Issue 12 addresses plant procedures necessary for the annual radiation protection program review; the procedures will be in place by October 1, 1996.

Compliance Plan Issue 13 addresses posting of radioactive material areas. Areas which contain unlabelled but potentially radioactive containers will be posted by December 31, 1996. There are also leased areas which have not been radiologically characterized. The characterization and any necessary reposting will be complete by December 31, 1998

Compliance Plan Issue 7 addresses testing of HEPA filter systems required to control worker exposure and practice ALARA principles. These systems will be identified and retrofitted or replaced to allow for adequate testing by June 30, 1997.

7.6 Conclusion

The NRC staff finds that the radiation protection program and associated Compliance Plan proposed by USEC provides reasonable assurance of adequate safety and are acceptable.

The regulations at 10 CFR §76.87(c)(3) require the TSRs to address criticality prevention. In addition, 10 CFR §76.89 requires USEC to maintain and operate a criticality monitoring and audible alarm system. The nuclear criticality safety program is also part of the management controls and oversight necessary to protect the public health and safety required by 10 CFR §76.35(a)(7). The nuclear criticality safety (NCS) program is described in §5.2 of the SAR and required by TSR 3.11.

TSR 3.11 establishes the foundation for the NCS program. USEC commits to establish, implement, and maintain the program as described in the SAR. The TSR further requires the NCS program to address the following elements: adherence with ANSI/ANS standards, NCS responsibilities, process evaluation and approval, design philosophy and review, criticality accident alarm system coverage, procedure requirements, posting and labeling requirements, change control, operation surveillance and assessment, and technical aspects. The TSR requires all operations involving uranium enriched to 1.0 w/o or higher and 15 g or more of ^{235}U to be based on a documented NCS evaluation and to be performed in accordance with a NCS approval. The TSR sets the minimum margin of subcriticality of 0.02 in k_{eff} and a k_{eff} of $\leq$ 0.9634 (including the bias, uncertainty, and the margin of subcriticality) for all criticality calculations. The TSR further requires the double contingency principle to be used as the basis for design and operation of processes using fissionable materials; for those instances where double contingency is not met, TSRs shall be established, implemented, and maintained to prevent criticality from occurring. The staff concludes that this TSR sets an acceptable foundation for the NCS program.

8.1 NCS Administrative Requirements

The administrative aspects of the NCS program are discussed below. PORTS has committed to ANSI/ANS 8.1-1983, ANSI/ANS 8/7-1975, and ANSI/ANS 8.19-1984.

8.1.1 NCS Organization and Responsibilities

The General Manager has overall responsibility for NCS and approves the implementation of Nuclear Criticality Safety Approvals (NCSAs). The General Manager assigns responsibility and delegates commensurate authority to all levels of management for the implementation of the requirements contained in the approvals for these same operations. First-line managers are responsible, in their respective operations, for ensuring that personnel are made aware of the requirements and limitations established by NCSAs either through pre-job briefings, required reading, and/or training (based on the complexity of the change). These same managers are responsible for ensuring any new fissile material operations which do not have approved NCSAs will not be performed until the necessary approvals have been obtained.

Managers are trained in NCS and ensure all appropriate personnel (i.e., fissile material handlers) receive training as specified in the NCS procedure. This training provides personnel with the knowledge necessary to fulfill their respective NCS responsibilities. The fissile material operators are responsible for conducting operations in a safe manner in compliance with operating procedures and are required to stop operations if unsafe

conditions exist. The manager of NCS is responsible for the administration of the NCS program. This includes reviewing the overall effectiveness of the NCS program ensuring that NCS staff members are placed, trained, and qualified in accordance with written procedures and that NCS evaluations and NCS approvals are prepared and technically reviewed by qualified NCS engineers.

The NCS Section is independent of organizational elements performing operations which require NCS evaluation.

Qualified NCS and senior NCS engineers are responsible for performing the following functions: (1) providing NCSAs for fissile material operations, (2) performing walk-throughs of facilities which handle fissile material and advising appropriate supervision of any NCS concerns, (3) participating in investigation of incidents involving NCS and in the determination of recommendations for eliminating such incidents, (4) assisting in plant emergency preparedness planning, (5) providing support to the PORC and (6) participating in the review of procedures which involve fissile material operations to verify NCSA commitments have been effectively incorporated into operating procedures. As does any employee, the NCS Section personnel have the authority to halt any unsafe activity.

The responsibilities of senior NCS engineers performing technical reviews of Nuclear Criticality Safety Evaluations (NCSEs) are specified in the NCSE procedure. These responsibilities include verifying that sufficient information is documented to allow independent analysis, verifying that credible process upsets related to criticality safety are properly identified and evaluated, verifying compliance with the double contingency principle, checking for accuracy, and verifying applicability of the calculational methods.

8.1.2 Process Evaluation and Approval

NCS program responsibilities are implemented by internal policies and procedures that are controlled by a formal document control process for development, revision, documentation, review, and approval.

Each operation involving uranium enriched to 1 weight percent or higher ^{235}U and 15 g or more of ^{235}U is evaluated for NCS prior to initiation. The operation and related NCS requirements are documented in Part A and Part B of an NCSA. The evaluation is documented in a NCSE. The evaluation and approval process is governed by written procedures.

When a NCSA is needed for a particular operation, the organization responsible for performing the operations completes a NCSA request form (i.e., Part A Request for Criticality Safety Evaluation). Part A of the NCSA documents the operating organization's request for NCS evaluation and the description of the operation. This form is approved and signed by the manager of the operating group/section or his/her designee. The form is then submitted to the NCS Section for analysis.

In response to the request, an NCSE is prepared to document the analyses performed as specified in the NCSE procedure. Techniques such as an NCS parameter checklist, What-If analysis, or Hazard and Operability Analysis are used to identify and document potential

upset conditions presenting NCS concerns. The method selected is based on the complexity of the operation being considered.

Engineering judgement of both the analyst and the technical reviewer is used to ascertain independence of events and their likelihood or credibility. The basis for this judgement is documented in the NCSEs. Depending on the complexity of the operation, analytical methods such as Fault Tree and Event Tree Analyses are used in the evaluation process to examine potential accident scenarios. When the determination of the likelihood of an event is questioned, qualitative or quantitative estimates of event frequency are developed to support the determination.

The NCS evaluation process involves: (1) a review of the proposed operation and procedures, (2) discussions with the subject matter experts to determine the credible process upsets which need to be considered, (3) development of the controls necessary to meet the double contingency principle, and (4) identification of the assumptions and equipment (i.e., physical controls) needed to ensure criticality safety.

Once the NCSE is completed, a technical review of the evaluation is performed and documented. The NCS staff member who performs the technical reviews of NCS evaluations is a senior NCS engineer who has successfully met the requirements specified in the NCS qualification procedure. The minimum requirements for a qualified senior NCS engineer and a qualified NCS engineer are specified in the application and are appropriate.

Part B of the NCSA is prepared based on the results of the NCSE, and documents the conditions of approval (i.e., NCS requirements) for the operation. The conditions of approval include the process conditions which must be maintained to meet the double contingency principle or preserve the documented basis for criticality safety and restrict the modes of operation to those which have been analyzed in the NCSE. This may include development of a TSR. The requirements to be included in operating procedures and postings are identified.

The NCSA approval process first involves the acceptance of the NCSE and NCSA by the technical reviewer. A 10 CFR §76.68 review will then be performed as described in Section 6.3.2 of the application to determine whether prior NRC approval for the NCSA is required. If NRC approval is not required, the NCSE and NCSA are reviewed by the PORC and, if acceptable, approved by the General Manager. The PORC reviews the NCSE and NCSA to verify technical accuracy, ensure all credible process upsets have been identified and ensure consistency with other NCSAs and other potentially conflicting requirements or regulations such as OSHA and/or radiation protection requirements. Once approved by the General Manager, the NCS controls, limits, evaluation assumptions, and safety items are verified to be fully implemented in the field. This verification process is performed by the operations organization and NCS personnel. The documentation of this verification process is maintained as a quality record along with the NCS evaluation. The manager or designee of the operating group or section then signs Part B, indicating acknowledgement and agreement with the limits and controls specified. The NCSA is then issued as a permanent or temporary document.

First-line management is responsible for implementing the conditions delineated in the NCSAs through the use of such tools as training, operating procedures, postings, and labels. First-line management ensures postings and labels are prepared and verifies that they are properly installed as required by the NCSA. The operating procedures are prepared or modified by first-line management to incorporate the NCSA requirements. First-line management is responsible for ensuring the employees understand both the procedures and NCSA requirements before the work begins.

Each completed NCSA is issued as a controlled document. The permanent NCSAs are maintained in a controlled manual which is issued to the personnel who need access to the NCSAs. The temporary NCSAs are issued to the appropriate personnel performing the temporary operation. Completed NCSEs and NCSAs are archived in fire proof safes maintained by the NCS Section Manager and are retrievable as permanent quality records in accordance with the plant records management system. The NCSA/NCSE process provides assurance that operations will remain subcritical under both normal and credible abnormal conditions. Any operations that do not comply with the double contingency principle are documented in NCSEs.

The double contingency principle as stated in ANSI/ANS-8.1-1984, Section 4.2.2, is as follows: "Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a critically accident is possible." The PORTS NCS program applies this principle by implementing controls either on two different parameters or by implementing two controls on one parameter. Controls include possible barriers (e.g., structures, vessels, piping, etc.); active engineered features (e.g., valves, thermocouple, flow meters, etc.); and administrative controls that require human actions to be taken in accordance with approved procedures. It two controls are implemented for one parameter the violation or failure scenarios of the controls shall be independent. Application of this principle ensures that no single credible event can result in an accidental criticality or that the occurrence of events necessary to result in a criticality is not credible.

There are three operations at PORTS which do not meet the double contingency principle. These are product cylinder operations, operation of the enrichment cascade, and removal of large cascade equipment (e.g., compressors, convertors, valves, etc). These operations have been evaluated to be safe and are described in the accident analysis section of the SAR. There are TSRs to ensure controls are in place for those operations identified above which do not meet double contingency. Sections 2.4 and 2.5 of the TSRs list controls for operation of the enrichment cascade and for removal and maintenance of enrichment cascade equipment, respectively. Section 2.3 of the TSRs provides the controls associated with ensuring moderation control for the product cylinders.

New operations and operations other than those identified as not meeting the double contingency principle shall comply with the double contingency principle. In the event future operations are found to not comply with the double contingency principle, the application (SAR section) will be modified to address this issue, and the modified application will be reviewed and approved by the NRC.

Emergencies arising from unforeseen circumstances can present the need for immediate action. If NCS expertise or guidance is needed immediately to avert the potential for a criticality accident, direction will be provided verbally or in writing. Such direction can include a stop work order or other appropriate instructions. A NCSA or other form of documentation will then be prepared to justify the actions taken once the emergency condition has been stabilized. This documentation shall be prepared within 48 hours following the stabilization of the emergency condition.

8.1.3 Design Philosophy and Review

Designs of new fissile material equipment and processes must be approved by the NCS Section before implementation and will include the use of favorable geometry or engineered controls on mass, moderation, volume, concentration, interaction, or neutron absorption, as the preferred approach over the issue of administrative controls. Advantage will be taken of the nuclear and physical characteristics of process equipment and materials provided control is exercised to maintain them.

The preferred design approach includes two goals. The first is to design equipment with NCS independent of the amount of internal moderation or fissile concentrations, the degree of interspersed moderation between units, the thickness of reflectors, the fissile material density, and the fissile material chemical form. The second is to minimize the possibility of accumulating fissile material in inaccessible locations and, where practical, to use favorable geometry for those inaccessible locations. Adherence to this approach is determined during the preparation and technical review of the NCS evaluation performed to support the equipment design. Fissile material equipment designs and modifications are reviewed to ensure that favorable geometry and engineered controls are used to advantage. Administrative limits and controls will be implemented in NCSAs to satisfy the double contingency principle for those cases where the preferred design approach cannot be met.

Fissile material equipment designs and modifications are reviewed to ensure that favorable geometry and engineered controls are used to advantage. Administrative limits and controls will be implemented in NCSAs to satisfy the double contingency principle for those cases where the preferred design approach cannot be met.

8.1.4 Procedure Requirements

Operations to which NCS pertains shall be governed by written procedures. These procedures contain the appropriate NCS controls for processing, storing, and handling of fissile material. The NCSA requirements which require employee actions shall be incorporated into the operating procedure. NCSA requirements are identified by placing a "commitment stamp" in the left hand margin next to the appropriate procedure step. Identifying these requirements in this way ensures changes to these requirements are not made without review and approval by the NCS Section. The NCSA requirements are incorporated into the appropriate procedures as required by the NCS procedure.

New and modified procedures are reviewed by the appropriate safety organizations, including the NCS Section as specified in the procedure for procedure control. The NCS Section reviews the procedures to verify that the appropriate NCSA requirements have

been incorporated and to verify that the proposed operation complies with NCS program requirements. The PORC must recommend approval of a procedure prior to issuance.

8.1.5 Posting and Labeling Requirements

NCS limits and controls for areas, equipment, and containers are presented through the use of postings and labels as specified in approved NCSAs and procedures. Postings and labels are proposed, reviewed, and approved during the NCSA review and approval process. These limits and controls are posted on the Nuclear Criticality Safety Requirements signs as required by the plant NCS procedure. The design of labels are prepared by the operating organization and concurred in by the NCS Section as specified by procedure. Approved NCSAs specify the wording for the postings. Limits and controls are printed in an appropriate size typeface and the postings and labels are placed in conspicuous locations determined by the line organization.

8.1.6 Change Control

Functional and physical characteristics of operations controlled for NCS are described in NCSAs and NCSEs. These components and features which are identified in the NCSAs and NCSEs are analyzed to determine the "boundary" of the system, encompassing those items that are essential to ensure operability. The boundaries are identified on system drawings and the configuration is verified to be as-built. These components and features are documented in a manual for each facility. Each time a change to a facility is planned this manual is reviewed, by the individual (e.g., design authority, systems engineer, operations supervisor, etc.) planning the change, to determine if the change affects SSCs relied on for safety. The design control manual specifies the organizations required to perform reviews of changes to SSCs. The required approvals are obtained before the change is implemented. The Configuration Control Board (CCB) verifies the required reviews have been performed before approval. If an item is relied on for the criticality safety of an operation, it will be identified through the work control process as an NCS SSC and NCS Section approval is required before implementing the change. The NCS Section reviews the NCSE for this specific operation and determines if the change affects the analysis performed and conclusions made in the NCSE. The change request will be approved by NCS only if the change does not impact NCS or once a revised NCSE has determined that the change is acceptable and meets NCS program requirements. In this way modifications to controlled operations are evaluated and approved prior to implementation. The items which require configuration change control are identified as specified by the safety system boundary identification procedure. These components and features are then reviewed during surveillances, assessments and walk-throughs of the facilities to verify that unauthorized modifications have not been made.

8.1.7 Audits, Inspections, and Surveillance

In order to ensure that the NCS program is properly established and implemented, USEC utilizes inspection, surveillances, assessments, audits, and walk-throughs.

The NCS Section performs internal surveillances of the plant's implementation of NCS limits and controls in accordance with procedural requirements, need, and results of

incident trending. The topics, scope, team members, and schedules for these surveillances are determined by the organization managers. Surveillance topics have included Criticality Accident Alarm Systems, Chemical Operations Facilities, Enrichment Cascade Facilities, Feed and Vaporization Facilities, Waste Management Activities, Product and Tails Withdrawal, NCS Validation of Computer Software, Laboratory Facilities, Maintenance Facilities, etc. Deficiencies identified during these surveillances are documented and reported to the appropriate levels of management in accordance with applicable procedures.

In addition to the NCS surveillances the operating organizations perform surveillances. At a minimum fissile material operations are reviewed for NCS on an annual basis. These surveillances are performed by the operating organization. NCS personnel are also present during these surveillances to provide NCS technical support. These surveillances include the inspection of facility modifications, operating procedures, compliance with NCSAs, postings, and waste generation and handling. These surveillances are performed as specified by the NCS procedure. The PORC provides review of the NCS Program.

Independent oversight is provided by internal audits of the NCS program conducted or coordinated by the USEC Safety, Safeguards, and Quality Manager. Internal audits of the NCS program shall be conducted annually. The purpose of these audits is to determine the adequacy of the overall NCS program. This includes the adequacy of the NCSEs, NCSAs, internal surveillances, and implementation of the NCS requirement.

NCS walk-throughs are performed by NCS personnel to determine the adequacy of implementation of NCS requirements and to verify that conditions have not been altered to adversely affect NCS. These walk-throughs are not performed based on an established frequency, but instead are performed as specified by the NCS oversight procedure. For example, a walk-through inspection can be performed in response to trend data, at the request of the operations personnel, or due to concerns raised by employees or the NCS staff.

The results of these surveillances, audits, and walk-throughs are documented and reported to appropriate managers. Identified deficiencies are documented and corrected according to the problem reporting system.

NCS deficiencies are recorded and the data trended to monitor and prevent future violations. Deficiencies are grouped into categories: building, organization/group, mass violation, volume violation, geometry violations, spacing violation, and unauthorized activities. Corrective actions are taken for adverse trends in accordance with the Quality Assurance Program.

8.1.8 Quality Assurance Program

A Quality Assurance Program has been written that sets forth the minimum quality requirements for all items, activities, and services provided at the PORTS site. USEC commits to the basic requirements and supplementary requirements provided in the ASME NQA-1, 1989 document.

USEC will apply quality assurance in a graded approach commensurate with the category of the SSCs and activities and services associated with the SSCs. A graded three-category quality assurance program for categorizing items and activities and to establish a level of quality in accordance with importance to safety has been provided. The levels established are as follows:

8.1.8.1 Category Q

Category Q applies to the safety systems described within the boundaries discussed in SAR Section 3.15. All QAP Sections 2.1 through 2.18 apply fully to Q items and activities.

8.1.8.2 Category AQ

Category AQ applies to SSCs within the boundaries discussed in SAR Section 3.15. The QAP applies to AQ items and activities to the extent described in Appendix A to the QAP.

8.1.8.3 Category NS

The remaining SSCs are categorized as non safety and this QAP does not apply to those items or services.

Appendix A of the QAP describes the extent to which Sections 2.1 through 2.18 apply to AQ items and activities or describes alternatives. Appendix A, Section 1. describes the quality program as it applies to Nuclear Criticality Safety. In Section 1 the applicant commits to applying all elements of the "Q" quality assurance program as described in Sections 2.1 through 2.18 of the QAP to NCS. As a result, the QAP for nuclear criticality safety is appropriate.

8.2 Criticality Accident Alarm System

A Criticality Accident Alarm System (CAAS) is provided to alert personnel if a criticality accident should occur. The system utilizes a distinctive audible signal to notify personnel in the affected area and initiate evacuation, thereby reducing personnel exposure to emitted radiation.

At PORTS, the CAAS detects neutron flux. The system uses clustered detectors. Each cluster contains three scintillation detectors. Activation of any two of the three detectors in a cluster will initiate evacuation alarms. The failure of any major component of the system will result in a notification that indicates the need for corrective maintenance.

The CAAS provides detection and alarm coverage for postulated criticality events that would produce an absorbed dose in soft tissue of 20 rad of combined neutron and gamma radiation at an unshielded distance of 2 m from the reacting material within 1 minute. The detection criteria are met by setting PORTS detectors at 5 milliroentgen per hour above the background radiation rate for the area(s) of coverage.

Operations involving fissile material are evaluated for NCS prior to initiation. The need for CAAS coverage is considered during the evaluation process. Coverage is provided unless it is determined that coverage is not required and that finding is documented in the NCSE. For example, areas containing no more than 700 g of ^{235}U or areas having material that is either packaged and stored in compliance with 10 CFR Part 71 or specifically exempt in accordance with 10 CFR §71.10, can be shown by evaluation not to require alarm coverage. Areas that do not contain any operations involving uranium enriched to 1 weight percent or higher ^{235}U and 15 g or more of ^{235}U do not require an NCSE and are not required to have CAAS coverage. 10 CFR §76.89(a) authorizes USEC to "describe for the approval of the Commission defined areas to be excluded from the monitoring requirement." This submittal to the NRC "must describe the measures that will be used to ensure against criticality including kinds and quantities of material that will be permitted and measures that will be used to control those kinds and quantities of material." In its letter dated May 22, 1996, USEC has submitted, in accordance with the Compliance Plan, Issue 11, a request for exceptions for CAAS coverage for certain areas of the facility as required by 10 CFR §76.89. The request is supported by an analysis. The request is still under review by the staff. The staff will complete its review during the transition period.

8.3 Technical Criteria

8.3.1 NCS Parameters

The use of NCS parameters (barriers) have been described in detail in the application. These parameters include moderation, volume, interactions, geometry, mass, enrichment, density, heterogeneity, concentration, reflection, and neutron absorption. Criteria have been established in the application to ensure appropriate control of each parameter identified with the exception of Enrichment and Neutron Absorption. An exception is the use of neutron absorbers. USEC has agreed that before a neutron absorber is used, for the purpose of complying with the double contingency principle, the details specifying the neutron absorber control program shall be submitted to the NRC for review.

Although the use of the NCS parameters has been described in detail for normal operations and activities, it is not clear how those parameters including Enrichment will be used for off-normal conditions. For example, the applicant has stated that small quantities of higher enriched uranium may be present on the plant site, within, or as samples from other facilities and in Department of Energy (DOE) operations described. However, it has been determined that the use of greater than small quantities of ^{235}U enriched to greater than 10.0 weight % is not discussed in the application except as it applies to DOE operations and it is known that certain routine operations may involve greater than 10.0 weight % enriched material. The applicant has not described or discussed actions that will be taken, if significant quantities of ^{235}U enriched to greater than 10.0 weight % are found in an unanalyzed or expected condition or location. The staff believes if USEC has determined that any of the NCS parameters described in the application, interim transitional requirements, NCSEs, NCSAs, or implementing procedures are not satisfied, USEC will treat this as an unreviewed safety question and actions to achieve and maintain adequate safety will be taken within a time period commensurate with the identified risk.

8.3.2 Methods of Calculation

NCS calculational methods have been derived and validated for the determination of criticality safety values associated with all process and storage operations performed at PORTS. To assist in these calculations, the applicant has validated computer codes, provided appropriate reference handbooks, and performs hand calculations in a prescribed manner.

For those cases where adequate references or experimental data are not available, NCS computational analyses are performed, which involve the calculation of k_{eff} to determine if the system will be subcritical under both normal and credible abnormal process conditions. Computer codes that simulate the behavior of neutrons in a process system or that solve the Boltzmann transport equation are used.

Computer calculations of k_{eff} provide a method to relate analytical models of specific system configurations to experimental data derived from critical experiments. A critical experiment is defined as a system which is intentionally constructed to achieve a self-sustaining neutron chain reaction or criticality. Critical experiments which have specific, well-defined parametric values and are adequately documented are termed benchmark experiments. Computer codes are validated using experimental data from benchmark experiments which, ideally, have geometries and material compositions similar to the systems being modeled. The PORTS computer code validation establishes the upper limit for calculations as $k_{eff} \leq 0.9605$ in accordance with the validation report POEF-T-3636, Revision 1, "Validation of Nuclear Criticality Safety, Software and 27 Energy Group ENDF/B-IV Cross Sections," dated January 1996. Reference to the minimum margin of subcriticality of 0.02 in k_{eff} plus the associated uncertainty and bias has been retained as a requirement and thus is consistent with $k_{eff} \leq 0.9605$ provided in the referenced validation report.

8.4 Compliance Plan Issues

The Compliance Plan for PORTS contains a description of those requirements within the application which currently are not being met (noncompliances), a justification for continued operations, a description of the actions taken to achieve compliance, and the schedule for completion of those actions. Noncompliances related to NCS are contained in the following Compliance Plan issues:

Issue 8	Nuclear Criticality Safety Approval Documents
Issue 9	Nuclear Criticality Safety Approval Implementation
Issue 10	Nuclear Criticality Safety Training for Managers
Issue 11	Exceptions for Criticality Accident Alarm System
Issue 23	Plant Changes and Configuration Management
Issue 24	Maintenance Program
Issue 26	Systems Approach to Training

Issue 30	Procedures Program
Issue 44	Criticality Alarms for Nearby Buildings

The staff finds these issues to be acceptable.

8.5 Technical Safety Requirements (TSRs)

The review of TSRs has been performed in a generic manner to ensure consistency from one issue to another and will not be addressed here.

8.6 Conclusions

Based on an in-depth review of the NCS program, the Compliance Plan and TSR requirements, it has been determined that PORTS will be operated in a safe manner. On-site reviews of each program element were conducted between November 1994 and April 1995 by knowledgeable NRC personnel. The staff finds the PORTS application regarding the NCS program to be acceptable.

The regulations in 10 CFR §76.60(d) require that USEC comply with the applicable provisions of 10 CFR Part 20. USEC describes its radiological environmental program in §5.1 of the SAR. TSR 3.16 requires USEC to establish, implement, and maintain the program described in the SAR. The radioactive waste management program is required by TSR 3.14. This chapter briefly discusses the USEC management of effluents and waste. USEC follows the ALARA principle as it pertains to releases. It is USEC policy to prevent or minimize the discharge of radioactive material to the environment. USEC has a pollution prevention program in place at the site.

9.1 Effluents

The environmental program includes a system of process and administrative controls to prevent releases above regulatory limits and to maintain effluents ALARA. Emission points and the emission controls are discussed in the following paragraphs.

PORTS has 13 emission points that are continuously monitored. The purge cascades are the primary airborne emission point. The purge cascades separate UF_6 from the light gases that have entered the process. The UF_6 is returned to the cascade and the light gases pass through chemical traps before sampling and venting to the atmosphere.

Exhaust from the cold recovery and wet air evacuation systems is passed through chemical traps prior to continuous sampling and venting. Each control area of the enrichment cascade has its own seal exhaust station. The seal exhaust stations collect and vent in-leaked air from inside the shaft seals of the cascade compressors. The air is sampled after passing through chemical traps, vacuum pumps, and mist eliminators. The X-344 sampling and transfer facility has a suction device that collects any small releases during cylinder disconnections. The exhaust is vented through a filter system prior to continuous sampling and venting.

There are also a number of maintenance and production activities using ventilation systems where the exhaust is vented through filter systems to control emissions. These vents are not routinely monitored since the emissions are low.

Action levels have been established for all monitored exhausts. Actions range from reviewing the data to closing the vent/stack.

The consequences of radionuclides released to the atmosphere from PORTS are determined by calculation of the committed effective dose equivalent (CEDE) to the maximally exposed person. The maximally exposed individual is located approximately 1770 meters east northeast of the plant site. The dose calculations are made using the CAP-88 package of computer codes. This is an accepted methodology. The 1994 dose to the maximally exposed individual due to airborne radionuclide effluents was 0.06 mrem. PORTS is required to submit an annual report to the Environmental Protection Agency that summarizes the airborne radionuclide emissions from the plant.

Wastewater streams at PORTS are discharged through eight USEC-leased site water outfalls. Most of the radiological waterborne discharges come from the decontamination and cleaning activities. The Environmental Compliance Status Report provides monitoring data for the outfalls.

Prior to discharge, decontamination and cleaning solutions are processed in the decontamination and recovery facility, along with uranium from other sources, such as the laboratory. Dissolved uranyl nitrate hexahydrate is separated by liquid-liquid extraction followed by calcination to U_3O_8. The remaining wastewater is treated for residuals and other heavy metals by pH adjustment and precipitation, for dissolved technetium by ion exchange, and for residual nitrates by biodenitrification. The treated wastewater is discharged to the on-site sewage treatment plant. Other radiologically contaminated process and cleaning wastewaters are collected, stored, and then treated using microfiltration and pressure filtration technology to remove radionuclides prior to discharge to the on-site sewage treatment plant. Sludge produced from the sewage treatment plant is drummed and stored pending future disposal. The liquid effluent is discharged to the Scioto River via a buried pipeline.

There are no public or known private drinking water intakes on the Scioto River or on Big Beaver Creek or Little Beaver Creek. However, USEC does calculate the CEDE for a hypothetical person. The calculated CEDE due to waterborne effluents during 1994 was 0.006 mrem.

The facility meets the dose limitations contained in the regulations. The PORTS effluent program meets the requirements of the regulations and is, therefore, acceptable.

9.2 Environmental Monitoring

PORTS has on-site and off-site permanent stations to collect ambient air samples. PORTS has three on-site stations, eight stations around the reservation boundary, and three off-site stations. Samples are analyzed monthly for gross alpha and beta deposition.

PORTS has an on-site meteorological tower located in the southern section of the site. It is equipped with instrument packages at the 10 m, 30 m, and 60 m levels to measure air temperature, relative humidity, and wind speed and direction. There is also ground level instrumentation to measure solar radiation, barometric pressure, precipitation, and soil temperature at 0.3 m and 0.6 m depths.

Thermoluminescent detectors (TLDs) are located both on and offsite. There are nine TLDs spaced around the perimeter of the plant, eight spaced around the reservation boundary, and two located offsite. The TLDs measure external gamma radiation and are collected and read on a quarterly basis.

Biological monitoring is conducted to assess the impact of plant operations on vegetation and animals in the vicinity of the plant. Vegetation is sampled semiannually from the site boundary and at various locations between the boundary and background locations. Samples of wide-blade grass are analyzed for total uranium concentration and ^{99}Tc beta

activity. PORTS also annually collects donated samples of food crops from local farmers and gardeners.

Soil and sediment samples are collected semiannually from around the area. Sediment and soil samples are analyzed for total uranium, gross alpha, gross beta, and technetium beta activity.

USEC collects water samples at the outfalls originating within the security area that discharge to waters of the United States. Outfalls with routine continuous flow are monitored with composite samplers and analyzed weekly. Grab samples are collected at the intermittent or rainfall outfalls. Samples are analyzed for total uranium, ^{99}Tc, gross alpha, and gross beta as described in the SAR. Grab samples are also collected both upstream and downstream in Little Beaver Creek, Big Beaver Creek, Big Run Creek, and the Scioto River.

A Program Coordinator is responsible for acquiring, storing, validation, trend analysis, and reporting of all field and analytical data from the monitoring program. Data is reviewed to identify possible trends and compared against background locations.

The monitoring program is acceptable to the staff.

9.3 Waste Management

The regulations at 10 CFR §76.35(m) require "A description of the program, as appropriate, for processing, management, and disposal of mixed and radioactive wastes and depleted uranium generated by operations." To meet this requirement, USEC submitted two plans: the Radioactive Waste Management Plan (RWMP) and the Depleted Uranium Management Plan.

The RWMP addresses the management of radioactive and mixed wastes. USEC projects that about 54,600 ft^3 of low-level radioactive waste is generated each year and about 8,300 ft^3 of mixed waste is generated. The RWMP describes the various wastestreams. Uranium is the primary radiological contaminant in the waste. The wastes may also contain trace quantities of transuranics (Np-237 and Pu-239) and fission products (Tc-99) from past processing of reactor return material. Each waste, whether liquid or solid, is characterized as releasable or not by the use of radiological surveys, sampling and analysis and/or the radiological status of its area of generation. Wastes not suitable for release are classified in accordance with 10 CFR §61.55 prior to disposal.

Wastes are collected at the source in drums, tanks, or boxes appropriate to the type of waste generated prior to treatment or transfer to a waste storage facility. Mixed wastes are transferred to a 90-day accumulation area equipped with secondary containment. After sampling, mixed wastes are transferred to DOE for storage until treatment/disposal options are available. USEC does not store mixed waste beyond 90 days. USEC has designated waste storage areas for different low-level wastes.

Radioactive wastes are containerized for shipment to storage or treatment facilities. Liquid wastes are processed in X-705 for removal of uranium from the wastewaters. The processed wastewater is discharged to the on-site sewage treatment plant.

The Depleted Uranium Management Plan describes USEC's program for the management and disposition of the depleted uranium produced at the facility. USEC has projected production levels of depleted uranium through the year 2005. Currently the cylinders containing the depleted uranium are stored on-site in cylinder storage yards. USEC conducts an initial inspection to document cylinder condition and then reinspects every 4 years to check for any indication of cylinder damage. Any damage discovered is evaluated for corrective actions which may involve more frequent inspection, cylinder repair, or emptying of the cylinder.

There are certain aspects of the Depleted Uranium Management Plan that are not currently in place; these are covered by Compliance Plan Issue 36. First, depleted uranium handling, moving, and stacking procedures are part of the procedure upgrade program and have not yet been updated. USEC will continue to utilize existing procedures until the upgraded procedures are in place at the end of 1996. The other item of noncompliance has to do with scheduling the follow-up inspections. Currently USEC does not does not conduct follow-up inspections. Followup inspections on the first USEC cylinders will not be due until July 1997. USEC will have the process for scheduling the followup inspections in place by December 31, 1996 which is before the first inspections begin; this is acceptable.

There are several potential uses for depleted uranium that are being investigated by both USEC and DOE. However, for planning purposes, USEC is assuming that the ultimate disposition strategy for the remaining inventory will consist of converting the solid to U_3O_8 using the pyrohydrolis process in which the UF_6 will be converted into a gas and combined with steam. The gas will react with the steam to form UO_2F_2 which is then converted to U_3O_8 in high temperature calciners. The U_3O_8 would then be packaged and shipped to an authorized repository. This approach is acceptable to the staff.

The USEC Radioactive Waste Management Plan and the Depleted Uranium Management Plan meet the requirements in the regulations.

The regulations at 10 CFR §76.35(g) requires USEC to submit a " compliance status report that includes the status of various State, local and Federal permits, licenses, approvals, and other entitlements, as described in §51.45(d) of this chapter. The report must include environmental and effluent monitoring data." As part of the application, USEC submitted an Environmental Compliance Report. The Environmental Compliance Report contained information on the environmental permits issued to the facility, including the principal permit limits, a summary of monitoring and emissions/effluent data for each permit, and a summary statement on the status of USEC compliance. USEC also provided a summary of the data from the environmental monitoring program. The report met the requirements of the regulation.

The regulations in 10 CFR §76.35(c) require the application to contain "any relevant information concerning deviations from the published Environmental Impact Statement, Environmental Assessments, or environmental permits from which the Commission can

prepare an environmental assessment related to the compliance plan." To meet this requirement USEC submitted a report called Supplemental Environmental Information. The information in this report and other information in the application was used in preparation of the environmental assessment that the staff prepared to support approval of the Compliance Plan. The report met the requirements of the regulation.

Chapter 10 CHEMICAL SAFETY

10.1 Regulatory Requirement

10 CFR §76.87(a) requires the Corporation to establish TSRs. The Corporation shall consider the analyses and results of the safety analysis report (SAR) submitted pursuant to Section 76.35. 10 CFR §76.87(c) requires that appropriate references to established procedures and/or equipment address each of the listed safety topics included in the TSRs. Chemical safety is one of the listed safety topics and includes the chemical hazards derived from radioactive materials, and plant conditions related to the hazards of chemicals on or near the site that may directly or indirectly affect radiation risk.

Examples of the chemical hazards derived from radioactive material processed at PORTS include: (1) the risk posed by radioactive materials that may be the cause of a severe chemical accident (radioactive materials that are explosive, flammable, highly reactive, or handled under high temperatures or pressures), (2) the risk posed by the chemical toxicity of some radioactive materials (the soluble forms of uranium, such as uranyl fluoride that results from the reaction of uranium hexafluoride with water), and (3) the risk to the environment or the public health and safety posed by a large accidental release of radioactive material with dangerous chemical properties (an accidental release of uranium hexafluoride would generate a plume of hydrofluoric acid when the uranium hexafluoride reacts with water vapor).

Examples of plant conditions that may either directly or indirectly affect radiation risk include: (1) the presence of hazardous chemicals or materials (flammable, pyrophoric or explosive) near radioactive materials or essential plant facilities for the control and containment of radioactive materials, and (2) the proximity of toxic chemicals or inert gases to essential areas of the PORTS site, such that an accidental release could render those areas uninhabitable.

10.2 Applicant's Chemical Safety Program

TSR 3.7.11, Chemical Safety Program, provides a clear commitment for the establishment, implementation, and maintenance of a chemical safety program, as described in Section 5.6 of the SAR.

Section 5.6 describes the chemical safety program at PORTS. The program integrates the environmental, safety, and health management systems. The chemical safety control strategy requires the identification and listing of chemicals in current use. The chemicals are categorized by potential chemical risk and nuclear safety significance, and are addressed by either Section 4 of the SAR Accident Analysis, the process safety management (PSM) program, or the industrial health and safety (IH&S) programs for chemical hazards. Existing plant programs are used to assure chemical safety through the incorporation of technical and administrative controls to manage risk. Those programs involve the development and use of operating procedures, site-wide safety procedures, operator training, maintenance, configuration management, emergency planning, incident investigation, audits and inspections, quality assurance, human factors, and detection and

monitoring. Cross-references are made to the program requirements addressed in other sections of the TSRs.

The potential for a release of hazardous chemicals and the impact on nuclear safety are analyzed in Section 4 of the SAR. The analysis considers both the radiological and toxicological effects of UF_6. The analysis recognizes the possible risks of a severe chemical accident associated with processing UF_6, and the chemical toxicity associated with the release of hydrofluoric acid and soluble forms of uranium that would result from the reaction of UF_6 and moisture in the air. The technical and administrative controls necessary to assure an acceptable level of safety for plant workers and members of the public from the processing of UF_6 are addressed in other sections of the Certification Evaluation Report.

Plant conditions that may either directly or indirectly affect radiation safety are addressed by one of two programs. A number of hazardous chemicals at PORTS are managed under the OSHA PSM program per 29 CFR §1910.119 to prevent an impact on nuclear safety, the workers, and members of the general public. Those chemicals are chlorine, chlorine trifluoride, hydrogen fluoride, and fluorine. The SSCs associated with those chemicals are classified using a graded approach as described in the QAP in order to assure proper configuration control, maintenance, and inspection. The uses, locations, and quantities of those chemicals are described in the SAR. A process hazards analysis (PHA) is required by the OSHA PSM rule to be performed for each of the chemical systems handling those hazardous chemicals.

Two NRC Headquarters team site visits were conducted on December 5-9, 1994, and January 30 through February 3, 1995, to assess the adequacy of the chemical safety program at PORTS, walkdown those hazardous chemical systems, and review the associated PHAs. Observations made during those site visits are documented in NRC Observation Report Nos. 70-7002/94002 and 70-7002/95001. The overall chemical safety program was in the process of being formalized at that time. The material condition of the hazardous chemical systems was found to be generally acceptable. The PHAs performed for the chlorine systems appeared to be thorough and rigorous and the PHAs for the remaining hazardous chemical systems were underway. Five of the nine required PHAs had been completed and the remainder are scheduled for completion by May 1997 (in accordance with the PSM rule).

The performance of a PHA provides an acceptable method of identifying potential hazards and recommendations for reducing the risks associated with the specified chemical systems. The OSHA PSM program at PORTS contains clear requirements that the PHA recommendations will be evaluated by management and that the recommendations accepted by management will be prioritized and performed according to an approved schedule. Completion of the PHAs in accordance with the schedule provided in keeping with the OSHA PSM program is acceptable because the remaining chemical systems are being operated and maintained in accordance with DOE-approved plant procedures and programs as required by Regulatory Oversight Agreement requirements 3.19.1 and 3.19.2.

Hazardous and toxic chemicals not covered by the above programs are managed by industrial hygiene and safety (IH&S) programs without regard to a threshold inventory barrier to enable the safe use of these chemicals. The chemical identification and inventory control feature involves three processes: (1) the baseline identification of chemical inventories used on site, (2) a formal engineering service order (ESO) program to address modifications to existing systems in order to consider new or revised chemical applications on site, and (3) contractor control to ensure that hazardous and toxic materials brought on site are properly authorized and controlled. These proposed administrative controls provide an acceptable level of assurance that chemicals used on site in less than bulk quantities will be identified, evaluated, and adequately controlled.

10.3 Compliance Plan Issues

USEC does not have in place all the aspects of a mechanical integrity program for covered hazardous chemicals to assure adequate confinement measures for those hazards. This is addressed in Issue 39 in the Compliance Plan. The mechanical integrity program for maintenance and inspection PSM requirements will be implemented by May 26, 1997, which is consistent with OSHA requirements. Until the program is complete, the facility will continue to use the program that existed under DOE regulatory authority. As an additional measure, PORTS will use work packages when performing maintenance on chemical systems. The staff finds the justification for continued operation and the schedule for completion to be acceptable. Issue 20 of the Compliance Plan covers the use of hazardous chemicals by DOE and other entities located on the PORTS site. Under this item DOE and third party tenants will provide USEC information on their hazardous chemical use. These parties will be required to inform USEC prior to bringing hazardous chemicals on site. The staff finds the schedule to be acceptable.

Chapter 11 FIRE PROTECTION

The regulations in 10 CFR §76.35(a)(6) require the SAR to include a "description of equipment and facilities which will be used by the Corporation to protect health and minimize danger to life or property" such as "fire protection systems." The PORTS fire protection program is described in §5.4 of the SAR. TSR 3.12 commits to the establishment, implementation, and maintenance of the fire protection program. The following paragraphs briefly describe the PORTS fire protection program.

The program is implemented by Fire Services (FS), which is headed by a management official who has been assigned the responsibility of the Authority Having Jurisdiction (AHJ). The AHJ is a qualified fire protection professional having a degree in engineering or a technical curriculum and at least 6 years of applicable experience. The FS is staffed with Fire Protection Engineers and Fire Officers. The PORC provides oversight and review of the activities of the FS and all fire safety-related issues. The PORC Chair has established a Fire Protection Subcommittee to assist in its responsibilities in the area.

The program includes performance of annual building surveys for the purpose of ensuring continued fire safety of the operations. This activity includes review of the fire safety features of the buildings, their occupancy, the hazards of the processes and storage in the buildings, and any other related issues. This is considered an important activity as the fire safety status of a building may change with time because of changes in the facility, processes, or management methods. A review of sample building surveys indicates a methodical effort applied to the task, and the surveys should be effective in timely detection of many safety problems.

The program includes inspection, testing, and maintenance of the fire protection equipment, which includes fire water systems, sprinkler systems, and a fire alarm system. The procedures and the testing and maintenance frequencies are according to those developed for a DOE-Oak Ridge program, which are based on applicable National Fire Protection Association (NFPA) codes. Major elements of the inspection program and associated frequencies are described in the SAR.

PORTS has a hot work permit system designed to control operations such as cutting, welding, and other hot work, which will be conducted in a manner consistent with industry fire prevention practices. Line Managers, who issue the permits, receive training on fire safety. FS is notified prior to the initial use of a permit; field surveillance of work is conducted during routine inspections.

The program also includes training of the plant fire department personnel who provide emergency response service and manual fire fighting capability. Training is based on national standard emergency response methodology and includes site-specific issues. Specific training activities include fire fighting, hazardous material response, confined space rescue, emergency medical response, radiological emergencies, and rescue. The fire-fighting personnel participate in the drills and exercises as specified in the emergency plan. The FS is also responsible for control of impairments of fire protection equipment, investigating fire incidents, and providing input to plant design and modifications.

11.1 Fire Protection Equipment

The facility is protected by two fire water systems: a High Pressure Fire Water System (HPFWS) that supplies the sprinkler systems in the process buildings and a low pressure Sanitary and Fire Water System, which supplies sprinkler systems in the other buildings and other water needs. Both have independent water supplies. The HPFWS is supplied by a 300,000-gallon overhead tank and surface reservoir having a capacity of 4.76 million gallons. Both systems have redundant electric and diesel-driven fire pumps. The fire main systems are fitted with sectional and post-indicator valves and hydrants.

The fire alarm system is activated by water-flow sensors in the sprinkler systems and fire detectors and other sensors to detect plant upsets. The alarm is transmitted to annunciators in both the Building X-1007 Fire Station and the plant central control room, which are manned around-the-clock.

The plant fire department is equipped with one 1000 gpm pumper, a "hazmat" truck, and other emergency response equipment. The installed fire suppression and the fire fighting equipment is adequate for the facility.

11.2 Building Construction

The process building construction comprises of structural steel frames, non-bearing walls, concrete floors, and built-up roofs. The roofs contain layers of bituminous combustible material, so that the buildings do not comply with the requirements of Type I construction as classified by NFPA 220, Types of Building Construction. However, all of the process buildings are fitted with automatic sprinkler systems, which compensate for the departure from the Type I classification in that the probability of a fire inside the building involving the roof is greatly minimized.

The ventilation system for the process buildings comprises forced draft fans drawing fresh air at the first floor air intakes and delivering into the second floor space. Air is withdrawn from the second floor space through the second floor deck by a duct system and induced draft fans, and returned to the atmosphere or partially recirculated within the building. The ventilation system handles air at a large volumetric rate, which is necessary for the particular application. This is a fire protection concern, since high ventilation rate may assist propagation of fire, especially since each floor of the process building is effectively one very large fire area without compartments. The sprinkler system reduces this risk to an acceptable level.

Tunnels connect the switch houses, process buildings, and central control facilities. The tunnels contain cables mainly for control and communications and some 440 V AC and 250 V DC power cables. All cables are insulated to 600 volts except for communication cables that are located in separate low voltage trays. However, all cables have neoprene or PVC jackets which are considered flame retardant. The tunnels do not have automatic fire suppression systems or fire detectors. All materials of construction are noncombustible, the trays have been maintained free of debris and/or combustibles, and transient combustible loadings are small. The only reasonable source of ignition is electrical in origin.

The process buildings being very large, the maximum egress path lengths, prescribed by NFPA 101, Life Safety Code, from most parts of the buildings are exceeded. In TSR 3.23, PORTS commits to identifying and marking emergency egress routes in process areas and maintaining them free of obstruction.

11.3 Process Fire Safety

The predominant process-related fire hazard arises from the possibility of a breach in the forced lubrication system for the compressors and other machinery and the resulting spill catching fire. The accident analysis section of the application recognizes the hazard. Apart from the lubricating oil, which is combustible, the average fire loading of the process buildings is light. An aqueous film-forming foam (AFFF) is the ideal suppression agent for an oil-spill fire. The facility fire department is capable of delivering AFFF from its pumpers. The buildings are fitted with automatic sprinkler systems, which also can help suppress and control propagation of an oil spill fire. The probability of an oil-spill fire originating in a cell housing, or propagating into it from outside, is a concern that was carefully considered. There is no automatic fire suppression system inside the cell housing. However, the combustible loading inside the cell housing is light and a detector installed in it is capable of detecting fire. The plant Fire Services would be alerted of a fire in the cell housing and timely intervening with manual fire-fighting equipment is possible.

In Compliance Plan Issue 18, USEC has committed to complete a combustible loading analysis for the process buildings by June 30, 1997. This schedule is acceptable to the staff.

11.4 Fire Hazard Analysis

The facility has building surveys performed on the major buildings. These surveys provide building descriptions, changes since the last updating of the survey, and descriptions of the hazards therein. The documents provide a baseline for fire hazard evaluation. PORTS evaluates fire hazards for the major buildings annually and documents the findings in the building surveys.

11.5 Pre-Fire Planning

An Emergency Plan implementing procedure requires Emergency Packets to be developed and updated annually by the facility custodian. These packets, in the facilities, contain information about the building, the layout, specific hazards, and other information applicable to the facility. PORTS has not updated the Emergency Packets recently; this issue is addressed in Compliance Plan Issue 18. Emergency Management will update these packets to reflect current facility configurations and conditions.

The staff concludes that the PORTS fire protection program is acceptable and meets the requirements of 10 CFR Part 76.

Chapter 12 EMERGENCY PREPAREDNESS

The regulations in 10 CFR §76.35(f) require USEC to submit an "emergency plan that meets the requirements of §76.91." Section 76.91 describes the type of information to be included in the Emergency Plan. USEC submitted an Emergency Plan for the PORTS with the application. The following paragraphs briefly describe the PORTS Emergency Plan.

12.1 Plant Description

Chapter 1 of the plan adequately describes the NRC-regulated activities and provides a description of the uranium enrichment process. The plan also specifies the substances associated with the enrichment process which could pose hazards if released to the environment and the locations where they are used or stored at the plant. The plan adequately describes the plant's main process buildings and the various other buildings located on the plant site, and the area near the plant, which is a rural low-population area.

12.2 Types of Accidents

Chapter 2 of the plan adequately gives a brief description of each type of accident that could result in consequences beyond the DOE reservation boundary which could possibly require protective action recommendations to off-site agencies. It lists quantities and locations of radioactive and hazardous materials located at the plant.

12.3 Classification of Accidents

Chapter 3 of the plan describes the plant's system for classifying emergencies as alerts or site area emergencies, and the action levels required by each. The plan requires an alert be declared if an incident had led or could lead to a release to the environment of radioactive or other hazardous material but is not expected to require a response by off-site response organizations. The plan requires a site area emergency be declared if an incident had led or could lead to a significant release to the environment of radioactive or other hazardous material and could require response by off-site organizations to protect persons off site. The definitions for alert and site area emergency are acceptable. Specific emergency action levels for classifying and declaring an emergency are in an Emergency Plan Implementing Procedure.

12.4 Detection of Accidents

Chapter 2 of the plan adequately identifies the means for detecting accidents or abnormal operating conditions in a timely manner. These include a criticality accident alarm system, UF_6 detecting equipment, a fire protection system consisting of automatic sprinkler systems and fire alarms, and various chemical detectors.

12.5 Mitigation of Consequences

Chapter 5 and 6 of the plan adequately describe the means for mitigating the consequences of an accident. The plant has systems and instrumentation available for

detecting abnormal operating conditions that could result in an emergency. Provisions and procedures are provided for evacuating personnel in the immediate incident area and controlling access to the surrounding accident vicinity. Mobile fire-fighting equipment is maintained on site to support fire fighting and back up the fixed fire suppression systems. Provisions and procedures are provided for treating and transporting injured workers, and providing protective action recommendations to local off-site officials in potentially affected off-site areas. Section 7.6 adequately describes the program for assuring emergency equipment is operable and properly stored and maintained.

12.6 Assessment of Releases

Sections 5.2 and 6.4 provide a description of the methods and equipment for assessing releases of radioactive or hazardous material. Assessment actions during an Alert include increased surveillance of plant instrumentation and visual observation of incident conditions, and monitoring event conditions for potential changes in the emergency classification level. Assessment actions during a Site Area Emergency include assessment of on-site and off-site exposures regularly to determine if and when on-site sheltering may be required. Additional activities can include performing continuing emergency assessments for mitigating events and protective actions on site based on on-scene and field monitoring results, release information, and meteorological conditions for radiological or hazardous material releases. Radiation detection equipment is used on site for normal and emergency response use and the plant also maintains emergency monitoring instrumentation for chemically toxic material releases. Personnel involved in an emergency submit urine samples for analyses if exposure to contamination is suspected. Depending on the meteorological conditions at the time of the incident and the location of the emergency, fence-line sampling may be conducted by monitoring team personnel.

12.7 Responsibilities

Chapter 4 of the plan provides an adequate description of the responsibilities of plant personnel during an emergency. During an emergency the General Manager is authorized to declare an emergency and initiate the appropriate response. The PSS assumes dual role as Crisis Manager (CM) and Incident Commander (IC) until the Emergency Operations Center (EOC) is activated, at which time the General Manager or designee assumes the role of CM. This is conducted by procedural checklists and, if possible, face-to-face briefings. The EOC is automatically activated for alerts and site area emergencies. Once the CM responsibilities have been transferred from the PSS to the General Manager, the PSS maintains responsibility of IC at the incident scene. Chapter 3 describes those responsible for notifying the NRC Operations Center immediately after notifying the off-site authorities but no later than one hour after the declaration of an emergency. Chapter 7 of the plan identifies the department responsible for maintaining and updating the plan.

12.8 Notification and Coordination

Chapter 3 of the plan provides a commitment to notify off-site authorities of an emergency. Chapter 4, section 4.3 of the plan describes provisions for requesting off-site assistance and Chapter 5, section 5.6 and 5.7 describe medical transportation and treatment of contaminated workers. Chapter 6 of the plan adequately describes the facilities and

equipment at the plant for mitigating emergencies. Facilities include an emergency operations facility, plant control facility, command post, emergency operations center, central alarm station, on-site medical facilities, and decontamination facilities. Equipment includes communications equipment, emergency monitoring equipment and a plant weather monitoring system.

12.9 Information to be Communicated

Section 3.3 of the plan describes the information to be communicated to off-site response organizations and the NRC during emergency notifications. The information communicated includes plant status conditions, radiological/hazardous materials release data, recommendations for protective actions for off-site response organizations, and other applicable emergency information.

12.10 Training

Section 7.2 of the plan describes the type of training that the general plant personnel, emergency response organization and support personnel, other DOE reservation personnel, and off-site emergency support organizations receive. Training records are retained to document readiness assurance. Nevertheless, some items of non-compliance exist with training. Tenant organizations have not previously been required to attend emergency preparedness General Employee Training. The methodology for such training has not been developed and implemented. However, tenant organizations have participated in informal training sessions, including safety awareness meetings and plant alarms familiarization. They have participated in the drill and exercise program during evacuation and accountability exercises. These informal training sessions have demonstrated an ability to protect them. Off-site response organizations have taken plant familiarization tours and have participated in drills and exercises. Based on this they have demonstrated the capability to respond to site emergencies. The NRC staff feels the level of skill and knowledge currently held by the off-site response organizations is adequate. Updated formal training will be developed and provided to them on the revised emergency plan and procedures by December 31, 1996. In addition, a formal emergency preparedness General Employee Training is being developed for tenant organizations. The completion date for this is September 30, 1996.

12.11 Procedures for Safe Shutdown and Recovery

Chapter 9 of the plan describe the means of restoring the facility to a safe condition after an accident. Recovery and restoration activities are conducted to maintain exposures as low as reasonably achievable. A recovery organization is established and managed by a Recovery Manager. This manager has overall responsibility for recovery activities which include checking safety equipment and restoring it to normal conditions.

12.12 Exercises

Section 7.3 of the plan adequately describes the provisions in place to conduct biennial exercises of the emergency plan. Off-site response organizations and the NRC are invited to observe or participate in these exercises. In addition to the biennial exercises the plant

conducts drills to test individual elements of the emergency plan. The Emergency Management Drill and Exercise Committee is responsible for scenario development, scheduling, and identifying participants and evaluators. Members of the emergency response organization are required to participate in drills and exercises. Formal critiques are conducted, deficiencies are identified, and corrective actions assigned and tracked through completion or implementation. Corrective actions are assigned to appropriate personnel with scheduled completion dates. Communications checks with off-site response organizations are conducted quarterly and telephone numbers are checked and updated.

12.13 Hazardous Chemicals

Chapter 10 of the plan states that the plant complies with the Emergency Planning and Community Right to Know Act. Plant procedures have been developed for hazardous materials releases that are not classified as emergencies to ensure that requirements of Superfund Amendments and Reauthorization Act of 1986, Title III are met. Material Safety Data Sheets are maintained in various areas throughout the plant.

12.14 Comment From Off-site Response Organizations

USEC submitted the letters they received from off-site response organizations commenting on the emergency plan. No significant problems with the plan were identified. One organization indicated that they could not transport contaminated individuals on their helicopter but otherwise would be able to provide transportation, if requested. The plant indicated that they had other ambulatory services available, including their own, to transport contaminated individuals, and that this was not a concern. Another organization expressed concern that the General Emergency category had been removed from the plan. The General Emergency category is not required for emergency planning purposes in Part 76.

Chapter 7 of the plan indicates any changes to the plan are communicated to the appropriate off-site response organizations. Letters of agreement with off-site response organizations are reviewed and updated every four years, or more frequently if needed.

12.15 Changes to Emergency Plan

The regulations allow USEC to make changes to the Emergency Plan if the changes do not decrease the effectiveness of the plan and the changes are provided to the NRC and affected off-site response organizations within six months of the change. In section 7.1 of the plan, USEC indicates that any changes that are made to the Emergency Plan that do not decrease the effectiveness of the plan will be furnished to the NRC and affected off-site response organizations within 6 months of the change.

The staff concludes that the Emergency Plan meets the requirements in 10 CFR §76.91, and is therefore acceptable.

Chapter 13 SAFEGUARDS AND SECURITY

13.1 Protection of Classified Matter

The applicant must comply with the requirements of Title 10 of the Code of Federal Regulations (CFR) Part 95 "Security Facility Approval And Safeguarding Of National Security Information And Restricted Data" in order to use, process, store, reproduce, transmit, transport or handle National Security Information (NSI) and/or Restricted Data (RD) in connection with NRC-related activities. Additionally, in December 1993, the Chairman of the NRC and the Secretary of Energy signed a Joint Statement of Understanding on implementing the Energy Policy Act provisions on the regulation of gaseous diffusion uranium enrichment plants. Paragraph No. 4 of the Joint Statement of Understanding states that "DOE will be responsible for the administrative determinations relating to granting, suspending, adjudicating, or denying a security clearance, and for reinvestigating an individual's background for continued access." The applicant must also comply with the guidelines set forth in the "Joint Statement of Understanding" between DOE and the NRC.

In September 1994, the NRC's Division of Security (SEC) accompanied DOE's Office of Safeguards and Security on a security inspection of PORTS. During the inspections, SEC was able to observe the plant's security program for the protection of classified matter. Other visits by SEC members were made to the plant in FY 94 as part of training courses being offered at the plant site and for general reviews of the plant's security program. In FY 95, SEC met with USEC staff on several occasions to provide guidance and comments on a draft security plan for the protection of classified matter at Portsmouth. On September 15, 1995, as part of the Certificate Application, the applicant submitted a security plan for the protection of classified matter at PORTS for formal review. Comments were provided to USEC on October 13, 1995, and November 27, 1995. During the period of October 16-20, 1995, representatives from NRC's Division of Security and NRC's Financial Management, Computer Security, and Administrative Support Staff conducted a verification visit at PORTS to confirm that their security plan, including computer security segment, accurately reflected the security program. The visit was favorable and only minor discrepancies were noted.

Conclusion:

The NRC staff reviewed the applicants latest response to NRC's comments on the security plan for the protection of classified matter, Group B sections, dated December 15, 1995, and found them to satisfy the requirements of 10 CFR Part 95.

The applicant has made commitments which meet the requirements of 10 CFR Part 95 by providing an acceptable security plan that establishes controls to ensure that classified matter is used, stored, processed, reproduced, transmitted, transported, and destroyed only under conditions that will provide adequate protection and prevent access by unauthorized persons.

Accordingly, the NRC staff concludes that the security plan for the protection of classified matter at PORTS, when implemented and verified by the Commission, is acceptable in meeting the requirements of 10 CFR Part 95.

13.2 Physical Protection

13.2.1 Regulatory Requirements

The NRC regulations state that the certificant must comply with the requirements of 10 CFR Part 76, Subpart E, which cites specific requirements within Parts 70, 73 and 74 for physical protection of Category I (Section 76.113), Category II (Part 76.115), and Category III special nuclear material (Part 76.117) at GDP fixed sites during transport.

The USEC application for certification states that the USEC gaseous diffusion plants are to use, produce, and transport only Category III special nuclear material of low strategic significance (SNM-LSS). At this site the DOE will retain ownership and responsibility for certain SNM, buildings, and areas as defined in the DOE Compliance Plan. DOE will continue to conduct operations and activities which may involve Category I, Category II, and Category III at these locations. These activities will be conducted under DOE regulatory authority and DOE Orders until these operations are terminated or the buildings and areas involved are turned over to USEC. The conditions for turnover of the buildings and areas to the USEC are defined in the DOE Compliance Plan and specify that only Category III SNM-LSS would be used, produced, or transported within buildings or areas turned over to USEC. USEC will not receive Category I or Category II materials from DOE and the certification will not authorize receipt of such material. At time of turnover to USEC, regulatory authority will be transferred from the DOE to the NRC.

Under the Energy Policy Act of 1992, the Corporation is the agent authorized to negotiate the purchase for the U. S. Government of all highly enriched uranium (HEU), uranium enriched to 20% or more of the uranium-235 isotope, made available by any State of the former Soviet Union under a government-to-government agreement. Although allowed to possess HEU under the Act, the Corporation chooses not to possess HEU. The Corporation chooses to possess low enriched uranium (LEU); namely, uranium enriched to less than 20% of the uranium-235 isotope that qualifies as special nuclear material of low strategic significance. Under the terms of the certificate, the Corporation is allowed to enrich uranium to an enrichment level of up to 9.999% at PORTS.

The NRC recognizes that the certificant may need or opt to engage in production or nonproduction activities that involve other categories of material. In that event, the certificant must apply for and be certified by the NRC as meeting the applicable safeguards regulations in accordance with the category of material that the Corporation seeks to access, use or possess.

The certificant must meet the general performance objectives of 10 CFR §73.67(a), submit a physical security plan per §73.67(c), and comply with the measures for physical protection of SNM-LSS as required by §73.67(f) at plant sites and (g) for SNM-LSS in-transit. Guidance for preparation of the Physical Protection Plans and the Transportation Protection Plans by USEC is provided in Nuclear Regulatory Commission Regulatory Guide

5.59, Revision 1, "Standard Format and Content for a Licensee Physical Security Plan for Protection of Special Nuclear Material of Moderate or Low Strategic Significance, February 1983.

13.2.2 Physical Protection at PORTS

To achieve the level of physical security for SNM-LSS at the facility necessary to meet the general performance objectives the certificant must comply with the requirements of §73.67(f), namely:

(i) Store or use the material only within a controlled access area;

(ii) Monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities;

(iii) Assure that a watchman or off-site response force will respond to all unauthorized penetrations or activities; and

(iv) Establish and maintain response procedures for dealing with threats of thefts or thefts of this material.

13.2.3 Transportation Protection

The certificant, as shipper or on delivery to a carrier for transport, of SNM-LSS must comply with the requirements of §73.67(g)(1), namely:

(i) Provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification;

(ii) Receive confirmation from the receiver prior to commencement of a planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges and specified mode of transport;

(iii) Transport the material in a tamper indicating sealed container;

(iv) Check the integrity of the containers and seals prior to shipment, and

(v) Arrange for the in-transit physical protection of the material in accordance with the requirements of §73.67(g)(3) of this part, unless the receiver is a licensee and has agreed in writing to arrange for the in-transit physical protection.

The certificant, in receiving quantities and types of SNM-LSS, must comply with the requirements of §73.67(g)(2), namely:

(i) Check the integrity of the containers and seals upon receipt of the shipment;

(ii) Notify the shipper of the receipt of the material as required in §70.54 of Part 70; and

(iii) Arrange for the in-transit physical protection of the material of low strategic significance in accordance with the requirements of §73.67 (g) unless the shipper is a licensee and has agreed in writing to arrange for the in-transit physical protection.

The certificant, either shipper or receiver, who arranges for the physical protection for transport of SNM-LSS while in transport or who takes delivery of such material free on board (f.o.b.) the point at which it is delivered to a carrier for transport must comply with the requirements of §73.67(g)(3), namely:

(i) Establish and maintain response procedures for dealing with threat of thefts of this material;

(ii) Make arrangements to be notified immediately of the arrival of the shipment at its destination, or any such shipment that is lost or unaccounted for after the estimated time or arrival at its destination; and

(iii) Conduct immediately a trace investigation of any shipment that is lost or unaccounted for after the estimated arrival time and notify the NRC Operations Center within one hour after the discovery of the loss of the shipment and within one hour of the recovery of or accounting for such lost shipment in accordance with the provisions of §73.71.

The certificant in exporting SNM-LSS must comply with the requirements of §73.67(g)(4), namely:

(i) Submit a security plan describing how the certificant will comply with the requirements of §73.67(f) and (g), as specified in §73.67(c); and

(ii) Comply with the requirements specified in §73.67(g)(1) and (g)(3), above.

The certificant in importing SNM-LSS must comply with the requirements of §73.67(5), namely:

(i) Submit a security plan describing how the certificant will comply with the requirements of §73.67 (f) and (g), specified in §73.67(c), and comply with the requirements specified in §73.67(g)(2) and (g)(3), above; and

(ii) Notify the person who delivered the material to a carrier for transport of the arrival of such material.

The certificant must also comply with the requirements for import and export of SNM-LSS in 10 CFR §§73.73 and 73.74, and other applicable regulations cited or referenced within the Code of Federal Regulations. Other referenced Parts include, but are not limited to, Parts 2, 19, 20, 21, 30, 40, 70, 71, 73, 74, and 75.

13.2.4 Certificant's Plan Commitments

The certificant has committed in the Physical Protection Plan and the Transportation Protection Plan for this GDP to implement procedures and measures to comply with the requirements of §73.67(f) and (g):

Storage and Use of Material

The entire facility is enclosed within a controlled access area bounded by a chain link fence. A clear zone surrounds the perimeter fencing. Access into the controlled access area is controlled and monitored by a security force and security patrols to allow entry of only authorized employees, visitors, delivery materials and products. There are procedures to control and account for SNM-LSS to assure this material is used and stored only within authorized areas and only by authorized individuals.

Access Controls

Access through the perimeter fencing into the controlled access area is channelled through personnel and vehicle gates. Personnel entry points and vehicle gates are manned and monitored by security personnel when in use. Access into the site is based upon established need and authorization. Uncleared visitors are escorted. Personnel and packages are searched for items which could be used for theft or sabotage. Access control procedures include search of suspiciously acting persons and of packages. There are procedures for control and accounting of SNM-LSS to assure that this material is used and stored only in authorized areas by authorized personnel in performance of their duties.

Detection of Unauthorized Penetrations or Activities

The certificant monitors the site perimeters, the controlled access areas, and storage areas by scheduled and by roving security patrols and from fixed posts at site gates.

Response to Unauthorized Penetrations or Activities

The certificant has a trained and exercised security force and implements procedures to detect and respond to unauthorized penetrations or activities. The security patrols provide a detection and assessment capability and form the primary response. The remaining on-duty security elements and, if necessary, local law enforcement authorities provide backup and support. Communication between and among these elements is established,

maintained, and exercised on a periodic basis. The certificant has established and maintains liaison with local law enforcement authorities.

Personnel Trustworthiness

The certificant requires that all personnel working at the facility be screened for personnel trustworthiness. Personnel are cleared at levels commensurate with their site access authorization and duties.

Advance Notification and Confirmation of Shipment

The certificant has and implements procedures for advance notification both prior to shipment and for confirmation after arrival at destination for each shipment of SNM-LSS.

Inspection of Shipments

The certificant has and implements procedures for use of containers, seals, and locks during transport and inspections prior to and on receipt of shipment of SNM-LSS.

In-transit Physical Protection and Response Procedures

The certificant is responsible prior to shipment to arrange or to assure by written agreement from the licensee (shipper or receiver) that appropriate measures for in-transit physical protection of SNM-LSS are in place; to establish and maintain response procedures for dealing with threats of theft or thefts of this material; to notify or make arrangements to be notified immediately upon arrival of a shipment at its destination or when a shipment is lost or unaccounted for after the estimated time of arrival at its destination; to initiate an immediate trace investigation of such lost or unaccounted for shipment; and to notify the NRC Operations Center within one hour after discovery of such loss or of recovery or accounting for such lost shipment.

13.2.5 Plan Commitments and Regulatory Requirements

The certificant has complied with the requirements of §73.67(f)(1) by providing procedures and measures for the storage and use of low enriched SNM only within a controlled access area.

The certificant has complied with the requirements of §73.67(f)(2) by providing procedures and measures for monitoring of the controlled access areas to detect unauthorized penetrations or activities.

The certificant has complied with the requirements of §73.67(f)(3) by establishing procedures and measures for response to unauthorized penetrations or activities using security patrols, local law enforcement, and a communication capability.

The certificant has complied with the requirements of §73.67(f)(4) by providing procedures and measures for response and maintaining those procedures and copies of superseded material for three years.

The certificant has complied with the requirements of §§73.67(g)(1)(i), (ii), (iii), and (iv) by providing procedures and measures for advance notification prior to shipment, for confirmation of arrival of a shipment, for inspection of containers and seals prior to shipment; and for use of tamper-indicating and sealed containers during transport.

The certificant has complied with the requirements of §§73.67(g)(2)(i) and (ii) by providing procedures and measures to check on the integrity of containers and seals on arrival of a shipment at destination and by notifying the shipper of receipt of the material.

The certificant has complied with the requirements of §§73 67(g)(1)(v), 73.67(g)(2)(iii), and 73.67(g)(3) by arranging for procedures and measures or by assuring by written agreement from the licensee (shipper or receiver) that provisions for in-transit physical protection have been made; by providing or verifying that response procedures have been established and are maintained by the party responsible for arrangements for physical protection of the shipment; by verifying that arrangements have been made for notification of the shipper immediately on arrival of the shipment at the destination, or for any such shipment that is lost or unaccounted for after the estimated time of arrival at its destination; by verifying that arrangements have been made to conduct a trace investigation of any shipment that is lost or unaccounted for after the estimated time of arrival; and by providing for notification of the NRC Operations Center within one hour after loss, recovery, or accounting for such lost shipment.

The certificant in the PORTS Transportation Plan and the PORTS Physical Protection Plan has committed to and has provided procedures and measures to meet requirements for physical protection and response for domestic, export, and import shipments of SNM-LSS both in-transit and at PORTS as specified in sections 73.67 (g)(4), 73.67(g)(5), 73.73 and 73.74, and other applicable regulations cited and referenced in the Code of Federal Regulations.

13.2.6 NRC Staff Recommendation on Certification

The certificant's Physical Protection Plan and Transportation Protection Plan for PORTS have been reviewed by the Nuclear Regulatory Commission staff. The procedures and measures outlined in these plans satisfy the performance objectives and system capabilities required by 10 CFR Part 76 and other applicable Parts cited and referenced therein.

The Physical Protection Plan and the Transportation Protection Plan for PORTS are acceptable and meet the Nuclear Regulatory Commission requirements for physical protection of SNM-LSS both at this site and during domestic, export, and import shipment to and from this site.

13.3 Material Control and Accounting

The regulation in 10 CFR Part 76, Subpart E requires USEC to meet specific requirements within Parts 70 and 74 for material control and accounting for Category I, Category II, and Category III SNM. The PORTS possession limits for SNM-LSS are such that only safeguards requirements for Category III SNM-LSS apply to USEC production activities at

this plant. Specifically, USEC must comply with the applicable requirements of 10 CFR §§70.51, 74.11, 74.13, 74.15, 74.17, 74.33, 74.81, and 74.82.

NRC recognizes that USEC may opt to engage in production or nonproduction activities that involve other categories of material. In such event, USEC must apply for and be certified by the NRC as meeting the applicable safeguard regulations in accordance with the category of material that USEC seeks to access, use, or possess.

As part of the application, USEC submitted the Fundamental Nuclear Material Control Plan (FNMC Plan) for PORTS. Because of the nature of this plan, it is not publicly available. The plan describes how PORTS will meet the material control and accounting requirements. The staff concludes that the FNMC Plan for PORTS satisfies the performance objectives and system capabilities required by the regulations; therefore, the plan is acceptable.

However, there are several aspects of the FNMC Plan that are not yet in place and are therefore addressed in the Appendix to the Compliance Plan. This portion of the Compliance Plan is not publicly available due to its nature. The Appendix discusses four issues related to material control and accounting. With regard to Issue A-4 of the Appendix titled, "Possession of Uranium Enriched to Greater Than 10% U235," USEC has committed to completing all HEU Suspension activities by September 30, 1996. Although the Compliance Plan provides for possible slippage of the above schedule, USEC has committed in its letter to DOE dated August 6, 1996 that any slippage will not go beyond the date when NRC assumes regulatory oversight for PORTS. Also, with regard to HEU refeed operations, it is the staff's understanding that the 45.0 and 47.5 kilogram ^{235}U action thresholds identified in the SAR Section 3.7 Appendix will be adjusted downward whenever the current 95% confidence level uncertainty of such ^{235}U quantity exceeds 5.00% relative. The magnitude of such downward adjustment will be sufficient to assure at least a 95% level of confidence that the actual ^{235}U quantity never exceeds 50.0 kilograms. Based on these understandings, the staff finds the plans and schedules to be acceptable and the justifications for continued operations contain adequate measures.

Chapter 14 DECOMMISSIONING

The regulations at 10 CFR §76.35(n) require that "a description of the funding program to be established to ensure that funds will be set aside and available for those aspects of the ultimate disposal of waste and depleted uranium, decontamination and decommissioning," which are the financial responsibility of USEC. The regulations further state that "The Corporation shall establish financial surety arrangements to ensure that sufficient funds will be available for the ultimate disposal of waste and depleted uranium, and decontamination and decommissioning activities which are the financial responsibility of the Corporation."

As part of the application, USEC submitted a Decommissioning Funding Program Description. It addresses the scope of USEC's financial responsibility for decommissioning, a cost estimate and basis, and the funding mechanism.

Under the lease agreement with DOE and the Energy Policy Act, USEC is not responsible for the decontamination and decommissioning of the leased premises. DOE retains responsibility for the decommissioning, including decommissioning of any capital improvements (i.e., new buildings or equipment). USEC is financially responsible for the disposal of low-level radioactive waste and mixed waste generated by USEC and for the disposition of the depleted uranium generated from the enrichment process.

Costs of disposal of low-level and mixed waste are considered a production cost, assuming the waste is removed from the site during the year generated. The waste generated during decommissioning activities will not be USEC's responsibility. Currently USEC has a backlog of waste stored at the site. To come up with a cost estimate for disposal of its low-level waste, USEC used weighted average cost based upon existing contract prices or upon prices from contracts being re-negotiated. USEC projects that its liability on September 30, 1996, will be \$1.3 million. This is based on the assumption that most of the backlog will be removed from the site as of that date. Cost estimates for disposal of the mixed waste are \$1.3 million. Again this is based on contract prices and the assumption that much of the waste will be disposed of prior to September 30, 1996. This is acceptable to the staff. USEC will re-evaluate its cost estimates in October of each year. If the backlog is not disposed of in the time frame currently expected, adjustments to the fund will be made at that time.

The disposition of the depleted uranium tails is the major cost factor for decommissioning costs. USEC has utilized the same cost basis for disposition of depleted uranium tails as Louisiana Energy Services (LES). The total unit cost for the disposition of depleted uranium is estimated to be approximately \$5.27 per kilogram of depleted uranium (\$4 per kilogram of uranium for conversion to U_3O_8, \$1 per kilogram of U_3O_8 for disposal, and \$.27 per kilogram for transportation.) Due to inflation, the average cost is \$5.30 per kilogram. The staff finds this estimate to be acceptable. The staff does caution USEC that the cost of disposition of the depleted uranium tails for LES is an issue before the Atomic Safety and Licensing Board; if the Board rules that a higher (or lower) cost should be imposed, the staff will impose the same cost on USEC. An additional factor in USEC's future liability is the fact that DOE will be responsible for the ultimate disposition of the depleted uranium; if DOE determines how it will disposition the tails, the staff will consider imposing that cost basis on USEC. USEC estimates that approximately 22,649 metric tons of depleted

uranium will be generated by PORTS operations from July 1, 1993 through September 30, 1996. The total cost would be $120.1 million.

USEC is currently a government corporation, and as such has submitted a statement of intent to assure funding for USEC's decommissioning liabilities. The USEC Board of Directors issued resolutions that stated that it was the Board's intent to have funds available when necessary for decommissioning activities. USEC will review the decommissioning cost estimates and associated funding levels in October of each year. Adjustments will be made as necessary.

USEC has agreed that at the time the Corporation is privatized it will submit an executed sinking fund arrangement, a standby trust agreement, and a payment surety bond in executed form for NRC review. This item is addressed in the Compliance Plan.

The staff concludes that USEC's Decommissioning Funding Plan meets the requirements of the regulation, and is therefore acceptable.

Chapter 15 COMPLIANCE PLAN

The regulations in 10 CFR §76.35(b) require USEC to submit a "plan prepared and approved by DOE for achieving compliance with respect to any areas of noncompliance with the NRC's regulation." The plan must include a description of the areas of noncompliance, a plan of action and schedule for achieving compliance, and a justification for continued operation. As part of the application, USEC submitted a Compliance Plan that was prepared and approved by DOE. Since these items are necessary for initial compliance, the items contained in the Compliance Plan will not be subject to the backfit provisions of §76.76.

The PORTS Compliance Plan contains 44 issues that had areas of noncompliance. An additional four issues (issues related to safeguards)are considered proprietary and are treated in an appendix not publicly available. Each issue contains sections on the requirements, what the commitment was, a description of the noncompliance, a justification for continued operation, and a plan of action and schedule. Most of the noncompliance issues are discussed in other sections of this CER; the issue of transition from DOE to NRC regulation and the issue of the SAR upgrade are discussed in the following paragraphs. Compliance Plan Issues 6, 10, 14, 15, 19, 33, and 35 have been completed by USEC; Issues 31 and 43 were deleted.

Compliance Plan Issue 1 addresses the transition from DOE regulation to NRC regulation. According to Issue 1, the transition to NRC regulatory oversight is scheduled to occur 120 days after the directors decision on the certification is issued. This will allow time for USEC to revise procedures and train the operators on the TSRs. However, by letter dated August 16, 1996, USEC has requested that the 120 day period be extended such that the transition period would end on March 3, 1997. The staff has no objection to the extension of the transition period. A listing of open commitments under DOE will be provided to the NRC on the date NRC assumes regulatory authority to assure the commitments are not lost in the transition. This approach and timing is acceptable to the staff.

In order to establish the transition period, and the effective date of the certificate, the staff recommends the following condition:

> This Certificate of Compliance shall become effective on March 3, 1997. The NRC shall assume regulatory jurisdiction from the Department of Energy at 12:01 AM on March 3, 1997.

Compliance Plan Issue 2 addresses the Safety Analysis Report Upgrade. The SAR submitted as part of the certification application is based, in part, on the SAR 85. The SAR 85 has a number of areas which need to be updated with respect to the description of hazards, description of plant SSCs, human activities, and supporting safety analyses, including the following: (1) the "as-exist" plant configuration does not match plant descriptions in the SAR 85; (2) assumptions used for the initiating events need to be reevaluated; (3) the expected response of SSCs to events may be different than previously assumed (e.g., response to seismic events); and (4) there are assumptions and differences in the accident scenarios that could affect the releases and consequences calculated in the SAR 85. DOE is in the process of upgrading the SAR 85.

The DOE site-wide safety analysis upgrade (SAR upgrade) was initiated to address the above deficiencies, changes to the plant configuration implemented since the previous analysis was performed, and the revised safety requirements issued since development of the SAR 85. From the comprehensive analysis of all credible initiating events based on "as-exist" plant configurations, and the consequences from these accidents, the SAR upgrade will provide more clearly the technical basis for safety boundaries (i.e., safety systems, equipment, components, etc.) and human activities relied upon to ensure safety. In addition, the SAR upgrade will provide descriptions of the various safety programs (based primarily from the certification application). This final "safety basis", derived from the SAR upgrade, will provide the necessary safety baseline from which future plant modifications can be made pursuant to 10 CFR §76.68. Since the SAR upgrade is required to achieve compliance with NRC requirements, any safety improvements called for by the SAR upgrade will not be subject to the backfit provisions in 10 CFR §76.76.

Pending completion of the SAR upgrade and achievement of compliance, the current safety basis for continued operation is the plants' implementation of, and adherence to, safety requirements in OSRs (Operation Safety Requirements) and plant procedures, developed over 100 years of combined operating experience for the 3 GDPs. This safety basis will continue to be utilized by the GDPs until completion of the SAR upgrade and implementation of the approved amendment request addressing results of the SAR upgrade. The TSRs, which will take effect upon NRC assuming regulatory oversight of the GDPs, are based on the safety requirements in OSRs and additional safety requirements clarified by the NRC staff during the certification process. Current safety procedures will remain in effect after NRC assumes regulatory oversight of the GDPs. These safety procedures can be changed only in accordance with the procedure TSR which provides the necessary safety assessment and both management and technical review before a procedure is changed or replaced. TSRs cannot be changed without prior NRC approval.

The staff concludes that the combination of the use of experienced plant personnel; use of the certification SAR; implementation of, and adherence to, TSRs (which supersede the OSRs); current as well as new GDP safety procedures; and commitments to compensatory measures and/or interim regulatory commitments will ensure that the GDPs continue to operate safely until the SAR upgrade and accompanying GDP modifications for safety are approved by the NRC and implemented at the GDPs.

The staff has reviewed all of the issues for the PORTS facility and concludes the actions, schedules and justifications for continued operation are acceptable and that the Compliance Plan should be approved.

Issuance of the Certificate of Compliance for operation of PORTS will not have a significant effect on the human environment. Regulation by the NRC will not result in any environmental impacts beyond those previously considered by DOE in its environmental reviews and which currently exist or would be expected to continue absent NRC regulatory oversight. Therefore, in accordance with 10 CFR §51.22(c)(19), neither an environmental assessment nor an environmental impact statement is warranted for the certification of the PORTS. This determination only applies to those aspects that are in compliance with 10 CFR Part 76.

An Environmental Assessment (EA) was prepared as part of the Compliance Plan and application review process. The EA concluded that the environmental effects of approving the Compliance Plan will be insignificant. The EA further concluded that the Compliance Plan is sufficient to ensure that, during the interim of noncompliance, plant operations related to areas of noncompliance will not significantly affect the quality of the human environment.

The EA resulted in a Finding of No Significant Impact (FONSI). The FONSI has been forwarded with the Notice of Decision for publication in the Federal Register.

Chapter 17 AUTHORIZATIONS AND EXEMPTIONS

USEC has requested authorization to release items for unrestricted use if the surface contamination is less than limits established in the SAR. The limits are consistent with those established in the NRC's April 1993 document entitled, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." This request is consistent with accepted industry practice and, therefore, approval of authorization is recommended.

USEC has requested an exemption from the requirement of 10 CFR §20.1904, "Labeling Containers," which requires that each container of certified material bears a durable, clearly visible label. In lieu of this requirement, USEC will post a sign stating that every container may contain radioactive material. When containers are moved from contaminated areas, a survey is performed to ensure that contamination is not spread. During movement of UF_6 cylinders, which are easily identifiable, radiological workers are in attendance. This exemption is consistent with accepted industry practice and, therefore, approval is recommended.

Chapter 18 TERM OF CERTIFICATE

The initial certificate will be issued for an affective period of approximately 2 years, with an expiration date of December 31, 1998. This is consistent with the new provision in Public Law 104-134, the USEC Privatization Act, which amended Section 1701(c)(2) of the Atomic Energy Act replacing the requirement for an annual application for a certificate of compliance with the requirement for an application to be filed "periodically, as determined by the Commission, but not less than every 5 years." The staff believes that 2 years is a reasonable period for the first certificate of compliance; in 2 years significant progress will occur in implementing plant improvements specified in the Compliance Plan.

The requirements in §76.31 and §76.36 for an annual application were based on the previous statutory requirement for an annual application, which has been superseded. Therefore, the exemptions from these requirements are justified under §76.23, which specifically allows the NRC to grant such exemptions from the requirements of Part 76 as it determines are authorized by law and will not endanger life, property, or the common defense, and are otherwise in the public interest. The exemptions meet these criteria.

Therefore to accommodate a 2-year certification period, the staff recommends the following condition to grant an exemption from the requirement to submit an annual application for certificate renewal in 1997:

> The United States Enrichment Corporation is hereby granted an exemption from the requirements in 10 CFR §§76.31 and 76.36 requiring submittal of a renewal application for the year 1997. The next renewal application shall be filed by April 15, 1998.

Chapter 19 CONCLUSIONS

Upon completing the compliance evaluation of USEC's application, including the SAR, TSRs, program plans and Compliance Plan, the staff concludes that there is reasonable assurance that the plant will continue to be operated such that public health and safety will be adequately protected, and that the common defense and security will not be endangered. Furthermore, the staff determined that the application fulfills the requirements of 10 CFR Part 76. The staff recommends that USEC be issued a Certificate of Compliance in accordance with statements and representations contained in the SAR, program plans, and TSRs. The staff further recommends approval of the Compliance Plan. The staff recommends that the following conditions be part of the certification:

> The United States Enrichment Corporation shall conduct its operations in accordance with the statements and representations contained in the Certification Application dated September 15, 1995, and revisions dated January 19, 1996, May 31, 1996, July 26, 1996, August 1, 1996 and August 12, 1996, and in the Compliance Plan submitted July 12, 1996, July 15, 1996, and July 18, 1996 and the revision submitted August 1, 1996.
>
> United States Enrichment Corporation shall conduct its operations in accordance with the Technical Safety Requirements that are contained in Volume 4, Revision 5 of the Application dated August 1, 1996, as modified by Revision 6 of the Application dated August 12, 1996. Changes to the Technical Safety Requirements shall require NRC approval prior to implementation.
>
> This Certificate of Compliance shall become effective on March 3, 1997. The NRC intends to assume regulatory jurisdiction from the Department of Energy at 12:01 AM on March 3, 1997. (Not withstanding any references in the Compliance Plan to other scheduled dates for assumption of jurisdiction.)
>
> The United States Enrichment Corporation is hereby granted the special authorizations and exemptions in Chapter 1, Section 1.8 of the Safety Analysis Report, Revision 5.
>
> The United States Enrichment corporation is hereby granted an exemption from the requirements in 10 CFR §§76.31 and 76.36 requiring submittal of a renewal application for the year 1997. The next renewal application shall be filed by April 15, 1998.

Chapter 20 ACRONYMS AND ABBREVIATIONS

ACR	area control room
AEC	Atomic Energy Commission
AFFF	aqueous film forming foam
AHJ	authority having jurisdiction
AIHA	American Industrial Hygiene Association
AIREK	a computer code for analyzing criticality safety
ALARA	as low as is reasonably achievable
ALI	annual limit of intake
AMAD	activity mean aerodynamic diameter
ANS	American Nuclear Society
ANSI	American National Standards Institute
AQ	augmented quality (items)
ARA	airborne radioactivity area
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing & Materials
CA	contamination area
CAAS	criticality accident alarm system
CADP	cascade automatic data processing system
CAM	continuous air monitor
CC	Cascade Coordinator
CCB	Change Control Board
CCZ	contamination control zone
CEDE	committed effective dose equivalent
CER	Compliance Evaluation Report
cfs	cubic feet per second
Ci	curie
CIP/CUP	Cascade Improvement Program/Cascade Uprating Program
Cl_2	molecular chlorine
ClF_3	chlorine trifluoride
CM	crisis management
D&D	decontamination and decommissioning
DAC	derived air concentration
DF	design feature
DOE	Department of Energy
DOP	dioctyl phthalate
dpm	disintegrations per minute
EA	Environmental Assessment
EBE	evaluation basis earthquake
EBS	evacuation booster station
EOC	Emergency Operations Center
EP	environmental protection
EPA	Environmental Protection Agency
ERC	Emergency Response Commander
ERP	extended range product
ERPG	emergency response planning guidelines
ESO	engineering service order

F/S	freezer/sublimer
F_2	molecular fluorine
FNMC	Fundamental Nuclear Material Contrc
FONSI	finding of no significant impact
fpm	feet per minute
FS	Fire Services
ft	feet
g	(1) acceleration of gravity, (2) gram
GCEP	Gaseous Centrifuge Enrichment Project
GDP	gaseous diffusion plant
GERT	general employee radiological training
gpm	gallons per minute
HCA	high contamination area
HEPA	high efficiency particulate air (filter)
HEU	High Enriched Uranium
HF	hydrogen fluoride
HGSYSTEM/UF_6	a suite of computer codes for calculating downwind airborne concentrations for UF_6 releases
HPFWS	high pressure fire water system
HRA	high radiation area
IAEA	International Atomic Energy Agency
IC	Incident Commander
ICRP	International Commission on Radiological Protection
IH	industrial hygiene
k_{eff}	criticality coefficient
LAW	low-assay withdrawal
lbs	pounds
LCO	limiting condition for operation
LCS	limiting control setting
LES	Louisiana Energy Services
LLD	lower limit of detection
LLRW	low-level radioactive waste
LMUS	Lockheed Martin Utility Services, Inc.
m^3	cubic meter
MAWP	maximum allowable working pressure
milli	an arithmetic multiplier equal to 10^{-3}
min	minute
MSL	mean sea level
MW	one million watts (megawatt)
NCS	nuclear criticality safety
NCSA	nuclear criticality safety approval
NCSE	nuclear criticality safety evaluation
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NQA	nuclear quality assurance
NRC	Nuclear Regulatory Commission
NS	(1) nuclear safety, (2) non-safety (item)
NSI	National Security Information

NUREG	any of numerous regulatory and technical reports issued by the NRC
OJT	on-the-job training
OSHA	Office of Safety and Health Administration
OSR	Operational Safety Requirement
p	an arithmetic multiplier equal to 10^{-12}
PCF	plant control facility
PGDP	Paducah Gaseous Diffusion Plant
pH	a symbol for the acidity or alkalinity of a solution
PHA	process hazards analysis
PMF	probable maximum flood
PNL	Pacific Northwest Laboratory
PORC	Plant Operations Review Committee
PORTS	Portsmouth Gaseous Diffusion Plant
psia	pounds pressure per square inch, absolute
psid	pounds per square inch, differential (pressure)
psig	pounds per square inch, gauge (pressure)
PSM	Program Safety Management
PSS	Plant Shift Superintendent
PW	product withdrawal
Q (list)	list of structures, systems, components, procedures, etc. crucial to safety and meeting the criteria of NQA-1
QA	quality assurance
QAP	Quality Assurance Program
RA	radiation area
rad	radiation absorbed dose
RCRA	Resource Conservation and Recovery Act
RCW	recirculating cooling water
RD	Restricted Data
rem	Roentgen equivalent man
RM	river mile
RMA	radioactive material area
RWMP	Radioactive Waste Management Plan
RWP	radiological work permit
RP	radiation protection
SAR	safety analysis report (part of the USEC application)
SAR 85	either of two safety analysis reports, one for each GDP, issued by DOE during and after 1985
SAT	systems approach to training
scfd	standard cubic feet per day
SL	safety limit
SNM	special nuclear material
SNM-LSS	special nuclear material of low strategic significance
SR	surveillance requirement
SSCs	structures, systems, and components
SWU	separative work unit
TEDE	total effective dose equivalent
TSR	Technical Safety Requirement
UF_6	uranium hexafluoride

UO_2F_2	uranyl fluoride
USEC	United States Enrichment Corporation
μ	an arithmetic multiplier equal to 10^{-6}

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U.S. Nuclear Regulatory Commission, "Standard Format and Content for a Physical Security Plan for Protection of Special Nuclear Material of Moderate or Low Strategic Significance," U.S. NRC Regulatory Guide 5.59 (Revision 1), U.S. NRC, Washington, DC, February 1983.

U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, U.S. NRC, Washington DC, November 1982.

U.S. Nuclear Regulatory Commission, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication," Regulatory Guide 8.24, U.S. NRC, Washington DC, October 1979.

Appendix A PUBLIC COMMENTS AND NRC STAFF RESPONSES

PUBLIC COMMENTS AND NRC STAFF RESPONSES

Introduction

On September 15, 1995, the NRC received an application for certification from the U.S. Enrichment Corporation (USEC) for the initial certification of the Paducah Gaseous Diffusion Plant (GDP) located near Paducah, Kentucky and the Portsmouth GDP located near Piketon, Ohio. A 45-day public comment period was provided which ended on November 6, 1995. On November 6, 1995 the NRC received from USEC a compliance plan addressing the areas at the plants that were not yet in full compliance with 10 CFR Part 76. The 45 day public comment period for the compliance plan ended on December 29, 1995. The NRC received a total of eleven (11) letters commenting on the application or compliance plan. Copies of the public comments received are available for public inspection and copying at the Commission's Public Document Room (PDR) in the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and in the Local Public Document Rooms (LPDR) established for these facilities. The LPDR for the Paducah plant is located at the Paducah Public Library, 555 Washington Street, Paducah, Kentucky 42003. The LPDR for the Portsmouth plant is located at the Portsmouth Public Library, 1220 Gallia Street, Portsmouth, Ohio 45662.

The NRC also held public meetings near each site concerning the application for certification and the compliance plans for the Portsmouth and Paducah GDPs. These meetings were held to solicit public input on the initial certification of these facilities. Transcripts of these meetings are also available at the PDR and LPDRs.

The following companies or individuals provided comments by letter or at the public meetings:

Comments Received by Letter

Central Midwest Interstate Low-Level Radioactive Waste Commission
Coalition for Health Concern
Mark Donham and Kristi Hanson
Environmental Protection Agency (EPA), Region 4
EPA, Region 5
Ronald Lamb
Nuclear Energy Institute (NEI)
Occupational Safety and Health Administration (OSHA), Cincinnati, Ohio
OSHA, Salt Lake City, Utah
Craig Rhodes
Velma M. Shearer

Comments Received at the Public Meetings

Portsmouth Public Meeting - November 28, 1995

Speakers:

Ms. Velma Shearer
Ms. Vina Colley
Mr. Tim Mitchell
Mr. Gerald Wilkins

Paducah Public Meeting - December 5, 1995

Speakers:

Ms. Jotilley Dortch
Mr. Ronald Lamb
Mr. A.B. Puckett
Mr. Mark Donham

The following is a listing of the comments received followed by the NRC staff response.

1.	Commenter:	Central Midwest Interstate Low-Level Radioactive Waste Commission
	Date of Letter:	December 19, 1995
	Reference:	Paducah Application

1.A. There are inconsistencies in the tables identifying low-level radioactive waste and mixed waste categories. USEC did not explain how they arrived at their projections for mixed waste, and their projections do not appear to include low-level radioactive waste generated as a result of management of depleted uranium or radioactive tails that are generated during the enrichment process.

Response:
The tables are not inconsistent. Nine waste streams are described as the principal streams; each of those waste streams consists of one or more waste categories. Although some individual categories have projected no generation of waste, all nine of the principal waste streams have an estimated volume. The basis for the 60,000 ft^3 and the 860 ft^3 estimates is contained in Tables 1 and 2. It is correct that the estimates do not contain the depleted UF_6 tails. The depleted tails are covered by the Depleted Uranium Management Plan.

1.B. The application does not indicate that information concerning off-site shipment is entered into the computer-based tracking system used for tracking movement of waste.

Response:
Waste is tracked from the point of origin to final disposal in the waste tracking system. Final treatment and/or disposal is entered into the database for the tracking system.

1.C. The application allows wastes to be released if the total activity is less than 30 pCi/g for either a solid or liquid.

Response:
This has been deleted from the application.

1.D. Is it correct that no treatment wastes are shipped directly to DOE facilities for disposal since section 6.6 indicates that wastes can be disposed of at DOE facilities?

Response:
Although treatment residues are not shipped directly to DOE for disposal, this does not prohibit disposal at DOE facilities.

1.E. The application does not provide information concerning disposal volumes or activities and does not indicate that disposal is in compliance with requirements.

Response:
Waste volumes are provided in Section 3. USEC is responsible for packaging and transporting the waste in accordance with the regulations and the requirements of the disposal site. The facility receiving the wastes for disposal is responsible for the actual disposal. Onsite disposal is not authorized in the NRC regulated areas.

1.F. It is not clear in the application who operates the waste tracking system. Also, not enough detail is provided concerning the identification number that is assigned to each waste container.

Response:
The Waste Management Group Manager is in charge of the waste tracking system. A unique identification number is assigned to the waste upon completion of the request for disposal form.

1.G. The Compliance Plan should indicate when USEC will begin implementing a quality control program. (This comment actually refers to the Radioactive Waste Management Program.)

Response:
The Radioactive Waste Management Program merely indicates that some areas of the plan are not being met. The specifics are contained in the Compliance Plan. The Compliance Plan contains the detailed information on the items of noncompliance; it includes a justification for continued operations and a plan of action that provides the completion dates.

2. Commenter: Coalition for Health Concern
   Date of Letter: December 28, 1995
   Reference: Paducah Application

2.A. The Paducah plant has contaminated the site and surrounding area and should no longer operate.

Response:
The Paducah plant is in compliance with applicable regulations and is operated in a manner that should assure the public health and safety and protection of the environment. Therefore, there is no reason to shut down the facility. The cleanup activities at the site will continue under DOE and will not be under NRC jurisdiction.

2.B. Seismic activity is minimized in the application.

Response:
USEC and DOE have identified weaknesses in the seismic analysis portion of the application. DOE is in the process of completing an upgraded Safety Analysis Report that will address these weaknesses. The Compliance Plan commits USEC to strengthen Paducah plant structures and increase the capacity to withstand a seismic event. Until the modifications are complete, USEC will be required to operate the Paducah plant at a reduced level of production that limits internal pressures to below atmospheric pressure. This reduced pressure will limit the possible consequences of a release of hazardous material in the event of an earthquake.

2.C. Nuclear criticality has not been adequately addressed.

Response:
The staff has reviewed the nuclear criticality safety program and found it to be acceptable. The commenter was not specific as to the exact nature of the concern.

2.D. Contamination from the contents of 6000 drums of radioactive waste has not been addressed.

Response:
It is not clear what waste the commenter is referring to; the staff assumes the commenter is referring to the depleted uranium tails cylinders that are stored at the site. DOE retains responsibility for all tails and for any other waste that was generated prior to July 1, 1993 when USEC took over operation of the enrichment facility. USEC has a radioactive waste management plan and a depleted uranium management plan that describes how the facility deals with its waste.

3. Commenter: Mark Donham
Kristi Hanson
Date of Letter: December 22, 1995
Reference: Paducah Application

3.A. Cumulative effects from past operations should be included in the accident assessment.

Response:
Cumulative effects from past operations are not part of an accident analysis. The primary hazard of this facility is the inadvertent release of UF_6; the pathway of concern is inhalation. Exposure due to accumulation in the environment would be very small.

3.B. Accident scenarios should be considered which might result in large releases of hazardous material.

Response:
The staff has reviewed the USEC accident analysis and found that it does adequately address potential accidental releases.

3.C. What is the plan for the 6,000 stored cylinders of depleted uranium?

Response:
Most of the cylinders stored at the facility are the responsibility of DOE; only the tails cylinders filled after July 1, 1993 are the responsibility of USEC. USEC plans to continue to store its cylinders at the site, pending development of a disposal solution by DOE.

3.D. The offsite risk of a criticality accident is not addressed.

Response:
The criticality safety program has been adequately described in the application. A criticality accident is not discussed in the context of an offsite analysis because the impact of a criticality is very localized; no significant impact would be expected offsite.

3.E. How much plutonium is at the plant?

Response:
At one time recycled uranium (containing some radioactive impurities) was used as feed material for the enrichment process. Contamination in piping and equipment still remains. The very small quantity of plutonium which exists as contamination, and 0.5 Ci as calibration sources and lab chemicals, are the only plutonium which USEC will be authorized to possess. Recycled uranium is no longer used as a feed material.

3.F. Liquid effluents are not adequately addressed.

Response:
USEC does sample the surface streams that receive effluent and runoff from the facility. The monitoring program is described in section 5.1 of the safety analysis report.

3.G. The earthquake risk is not adequately addressed.

Response:
See response to 2.B.

3.H. Information on production of depleted uranium should be made public.

Response:
The waste management plans are contained in volume 3 of the application and have not been withheld from public disclosure. The actual volume of tails generated was withheld from the original application at the request of USEC; this information is now publicly

available. The information on production of depleted uranium tails has been made publicly available in the Public Document Room, and the Local Public Document Room for the Paducah plant.

3.I. The emergency plan is inadequate. Are local authorities trained for nuclear situations? Are the hospitals equipped to handle radiation or chemical poisoning? Public warning sirens that can warn the public cannot perform this service, and there are no maps or evacuation routes depending on wind direction.

Response:
The emergency plan for the Paducah plant meets the requirements in 10 CFR 76.91, "Emergency planning." Offsite response organizations are invited to participate in the biennial exercises that test all or most of the basic elements within the emergency plan. If offsite fire fighting assistance is needed to fight a fire involving radiological/hazardous materials, knowledgeable members of the plant emergency response organization provide radiological/toxicological information and assistance. The hospitals that have agreed to provide assistance in the event of an emergency are equipped to handle contaminated injured individuals. The plant may also provide radiation protection personnel from the plant to assist the hospital. Both hospitals have Nuclear Medicine and Radiography departments. The existing public warning sirens do not provide total coverage of the immediate notification area; however, plans are in place to replace the existing sirens with a new siren system to ensure better coverage of the immediate notification area. The scheduled completion date for the new system to be installed and operational is March 15, 1997. An emergency broadcast system message is an alternate method to provide public warning and is currently available and functioning. It should be noted, however, that USEC is not required by Part 76.91, "Emergency planning," to have a public warning system and that the inclusion of the public warning system in the emergency plan for the Paducah plant was a voluntary decision made by USEC. No scenario has been identified that would require evacuation of members of the public.

4.	Commenter:	Environmental Protection Agency (EPA) Region 4, Atlanta, GA
	Date of Letter:	December 7, 1995
	Reference:	Paducah Application

Comment:
The application appears to reflect compliance status with the various permits and regulations under EPA jurisdiction.

Response: No response needed.

5. Commenter: Environmental Protection Agency
Region 5, Chicago, IL
Date of Letter: February 29, 1996
Reference: Portsmouth Application

5.A. The application is documented to meet the requirements under the Clean Air Act and National Emission Standards for Hazardous Air Pollutants for Radionuclides (NESHAPs). The Compliance Plan seems to satisfy the NESHAPs requirements.

Response: No response needed.

5.B. If it is true that the ambient air pressure is greater in the storage drum room than on the process operating floors, then any leak in the storage drum room could potentially leak to the lower pressure area of the process operating floors. This is not what Section 3.1.1.6.8.8, Secondary Confinement System, on page 3.1-78, seems to be stating.

Response:
It is true that in the event of a leak from a cylinder, there would be a tendency for the released gas to flow from the area of higher pressure (the storage drum room) to the area of lower pressure (the process operating floor). In the event of a leak, features to protect safety of the workers on the process floor include: (1) the drum storage room is well sealed; (2) an alarm would sound, alerting operators to prepare to move to a safe location; and (3) fans in the vicinity of the entrance to the drum storage room would tend to disperse the gas before it could escape from the drum storage room. The staff concludes that the combination of features provides adequate worker protection against harm from leaks in the drum storage room.

5.C. The first paragraph on page 3.1-87 appears to indicate that any slings with test dates less than one year old need certification, while those that would have tests dates over one year old were fine.

Response:
The commitment in the application is acceptable as written. The commenter misinterpreted the commitment.

6. Commenter: Ronald Lamb, Kevil, KY
Date of Letter: December 28, 1995
Reference: Paducah Application

6.A. The Paducah plant does not have a sufficient public warning system nor an effective evacuation plan.

Response:
See response to 3.I.

6.B. In a catastrophic emergency there may not be sufficient electricity to safely shut the operation down which could possibly lead to a criticality.

Response:
All plant structures, systems and equipment are designed to fail safely in the event of a loss of power and therefore would present no immediate hazard. There would be no out leakage of UF_6 gas because UF_6 gas in the diffusion process would return to a subatmospheric pressure and a criticality is very unlikely under those process conditions. Neither the Portsmouth or Paducah plants have ever sustained a complete loss of offsite power.

6.C. There is an unresolved safety question regarding seismic risk.

Response:
See response to 2.B.

6.D. NRC is understaffed at the plant and civil penalties should be levied.

Response:
Two resident inspectors onsite is consistent with what NRC has at nuclear power plants. There are also inspectors at the NRC regional office and at headquarters that inspect the facility periodically. Originally the NRC did not have the authority to levy fines, however on April 26, 1996, President Clinton signed into law H.R. 3019, legislation which includes a subchapter entitled the "USEC Privatization Act." This legislation includes several provisions affecting regulation of USEC by NRC. One of these provides authorization for NRC to impose civil penalties on USEC or its successor for failure to comply with regulatory requirements governing the gaseous diffusion plants.

6.E. USEC should not be allowed to keep the amount of waste produced at the plant confidential.

Response:
See response to 3.H.

7. Commenter: Nuclear Energy Institute (NEI)
   Date of Letter: November 6, 1995
   Reference: Paducah and Portsmouth Applications

Comment:
NRC's regulatory approach should be performance-based rather than prescriptive. Some of the NRC questions submitted to USEC indicate an overly prescriptive approach.

Response:
The staff agrees that, to the extent feasible, performance-based regulation is appropriate. However, for these plants, which have been operating for many years, the safety basis for operation is not well-defined in all cases. Therefore, the staff has required USEC commitments to many existing safety procedures established over time by the Department of Energy. This has resulted in many prescriptive requirements, but the staff believes that they are necessary to assure safety.

8. Commenter: Occupational Safety and Health Administration
Date of Letter: November 24, 1995
Reference: Paducah and Portsmouth Applications

8.A. When design changes affect general operating procedures, employees should be apprised of the changes and related training should be updated.

Response:
This is currently included in the training program in SAR Chapter 6.6.

8.B. Threshold limits in 29 CFR 1910.119 for hazardous materials should be adopted.

Response:
NRC does not have the authority to require USEC to comply with OSHA regulations.

8.C. Reference Paducah SAR, Emergency Plan, Page 1-3. Have less hazardous materials other than chlorine been considered for water treatment process, such as hypochlorite solution or ozone?

Response:
Other water treatment process has been considered. A bromate treatment process is being tested at Portsmouth. Depending upon its result, this process might be applied to Paducah.

8.D. Reference Paducah Quality Assurance Program, section 2.7.3.6. For items which were obtained from an internal parts or equipment source, preinstallation inspection should be required.

Response: The "preinstallation" inspection is part of the receiving inspection.

8.E. Reference Paducah Technical Safety Requirement (TSR), section 1.2.4. The statement "No training is required to perform the fire patrol" should be deleted.

Response:
USEC has deleted this statement.

8.F. Reference Paducah TSR, section 2.1.2.2. There is a potential for cylinder rupture during movement or handling, such as rolling, tilting, or suspending. Therefore, the first sentence is not inclusive and should be amended.

Response:
The statement applies to cylinders that are not being moved, and is appropriate in that context.

8.G. Reference Paducah TSR, section 2.2.3.1, Basis. The Basis states that the test parameters were approved in December, 1980. The safety committee should update or reevaluate test parameters.

Response:
USEC has revised and updated the Basis statement.

8.H. Reference Paducah TSR, section 2.3.3.1, Basis. Is there a pressure-limiting cut-off switch or sensor near the discharge expansion joint that would avoid the over-pressurization possibility which is described?

Response:
As described in Basis, dual high-pressure shutdown instrumentation is installed on the pump discharge pipe to prevent this type of accident.

8.I. In Paducah TSR, section 2.3.4.1, Condition C., a continuous smoke watch is initiated. It also should describe actions to be taken if smoke is observed.

Response:
Actions to be taken when smoke is observed are described in detail in plant implementing procedures.

8.J. Reference Paducah TSR, section 2.3.4.1, Conditions G and H. The action instructions are incomplete.

Response: USEC has revised and completed the instructions.

8.K. Unplanned medical treatment at a medical facility of an individual is described in Paducah TSR Section 3.8. It should also describe what capability is available if a larger number of people need to be treated.

Response:
This section of the TSR describes special reporting requirements. An event that requires unplanned medical treatment at a medical facility of an individual with radioactive contamination on the individual's clothing or body must be reported to the NRC. USEC is not required to describe medical capabilities for different situations.

8.L. There is lack of employees' involvement or participation in safety and health processes of various program developments.

Response:
Many of the programs (Nuclear Safety Programs, Quality Assurance Program, etc.) described in the application are part of USEC's management control program. Participation and involvement in these programs require some degree of experience and expertise which may not be possessed by non-supervisory employees. While program development is a management responsibility, it is anticipated that employees from all segments of the plant staff will be involved in program development as appropriate. The As Low As Reasonably Achievable (ALARA) subcommittee has a representative from the Oil, Chemical & Atomic workers bargaining unit.

9. Commenter: Occupational Safety and Health Administration
Date of Letter: March 8, 1996
Reference: Paducah and Portsmouth Compliance Plans

9.A. Describe distinctions between small fires and major fires in order to know how to respond to each.

Response: This section describes available features for responding to small and major fires. Distinctions for the purpose of responding to fires are not necessary here.

9.B. Is there a procedure for closing out noncompliance items completed?

Response:
For completed Compliance Plan (CP) items, DOE will conduct closure inspections to ensure that each item has been completed as committed. DOE's findings will be documented in its Site Safety Representative's inspection reports, and the CP will be updated to reflect the proper completion/closure. This procedure is documented in a letter from DOE to USEC, dated May 22, 1996.

9.C. Reference Paducah CP Issue 6, Commitments, 5.2.2.6 Procedure Requirements. Operation procedures should be written to cover not only normal, emergency, and temporary operations but also for start up following a turnaround and after an emergency shutdown.

Response:
The section to which the comment applies is limited to nuclear criticality controls that are contained in procedures. Operational procedures are covered in other areas of the application. USEC does have emergency procedures.

9.D. Reference Paducah CP Issue 10/Portsmouth CP Issue 13, Commitments, item 2. The statement "Containers located within radiological areas are not required to be labeled." should be explained further even when considering item 3 and Description of Noncompliance.

Response:
Section 5.3.1.7 of the application states that containers located in Radiological Areas within USEC leased areas are not individually labeled but instead are posted with signs that state any container may contain radioactive material.

9.E. Reference Portsmouth CP Issue 25, Justification for Continued Operation, In addition to normal operating procedures, emergency operating procedures, and others listed, current operations program should include emergency shutdown procedures and start up following a turnaround or emergency shutdown.

Response:
The Compliance Plan focuses on operating procedures. Emergency and shutdown procedures are covered by the application.

9.F. Reference Portsmouth CP. Is there involvement of employees in various safety related aspects of the operations including operating procedures, training, and emergency response?

Response:
Employees are involved in safety related aspects of operations. See also response to comment 8.L.

10.	Commenter:	Craig Rhodes, Brookport, Illinois
	Date of Letter:	November 14, 1995
	Reference:	Paducah Application

10.A. The assumptions regarding an earthquake risk at the Paducah plant are too low and should be more detailed.

Response:
See response to 2.B.

10.B. The plant lacks warning sirens in a two mile radius, there is no evacuation plan and the notification plan is inadequate.

Response:
See response to 3.I.

10.C. Hazardous material cleanup is inadequate.

Response: The Paducah Gaseous Diffusion Plant is regulated by EPA and must comply with applicable EPA regulations.

11.	Commenter:	Rev. Dr. Velma M. Shearer
	Date of Letter:	December 29, 1995
	Reference:	Portsmouth Compliance Plan

11.A. Containers with radiological contents should indicate the amount in the container.

Response:
In all restricted areas where containers may contain radioactive material, USEC is committed to posting caution signs stating that every container may contain radioactive material. USEC is also committed to provide appropriate safety training related to working in the vicinity and in handling of containers that may contain radioactive material, to all unescorted individuals entering restricted areas. In unrestricted areas, 10 CFR 20.1904 requires USEC to label all containers containing radioactive material with sufficient information, including the radionuclides present and their quantities, to permit workers to take precautions to minimize or avoid radiation exposures.

11.B. USEC should monitor individuals at shift closing and provide individuals with copies of dose monitoring results upon completion of reports.

Response:
USEC is required to provide radiation dose monitoring results to its workers and monitor them in accordance with 10 CFR 19 and 10 CFR 20, respectively.

11.C. Reference CP, Issue 21, "Management Controls," the Protection of national security interests is noted as a management responsibility. However, there is no indication as to who makes national security interest decisions. Will Lockheed-Martin Utility Services have jurisdiction to make political decisions regarding world-wide sales of enriched uranium?

Response:
USEC is bound by regulatory requirements to provide adequate security and safeguards for information, material, and equipment that requires protection to safeguard national security interests. Decisions regarding implementation of USEC's safeguards and security programs are made at appropriate levels in USEC's chain of command, in accordance with applicable NRC requirements. USEC is free, to the extent provided by law and regulation, to continue to provide enrichment services on a world-wide basis.

11.D. There is a comprehensive "flow-down" system of authority. However, no plan within the system is in place for a "flow-up" from any worker who may be able to contribute improvements to the enrichment or system process.

Response:
The CP issue that is the subject of this comment relates to having an appropriate organizational structure and effective management control to assure that all safety, safeguards and security requirements are met. This will include responding to safety problems identified by workers. It does not require a plan for use of beneficial ideas and suggestions from employees which are not related to safety.

11.E. A 500 mrem per year exposure limit for individual workers is recommended.

Response:
The annual whole body committed effective radiological dose limit contained in 10 CFR Part 20 for workers is 5 rem. In addition, 10 CFR Part 20 does not require monitoring if the annual dose is not expected to exceed 10 percent of the limit, which is 500 mrem. Also the radiological dose hazard at the Portsmouth plant is not significant. This is apparent based on monitoring data, for the years 1992, 1993 and 1994, during which a total of five individuals out of over 11,700 monitored for external radiation, and none out of 5,000 monitored for internal radiation at the Portsmouth plant, may have exceeded this monitoring threshold level.

11.F. Procedures need to be developed to allow workers to report such things as a faulty valve to prevent spillage or serious exposures.

Response:
There is a problem reporting program by which workers can report items like a faulty valve. Some of the minimum procedural requirements by which faulty valves to prevent spillage or serious exposures will be identified and reported are in the areas of internal audits and inspections, investigations and reporting, quality assurance, equipment control (lockout/tagout), preventive maintenance, etc.

11.G. The need for Atomic Vapor Laser Isotope Separation (AVLIS), which uses laser beams for the uranium enrichment process, at the Portsmouth plant is questionable.

Response:
The NRC has not received an application to license an AVLIS facility . This comment is outside the scope of the certification review.

Comments from Transcripts of Public Meetings

12. Portsmouth Public Meeting Transcript - November 28, 1995

Commenter: Ms. Velma Shearer

12.A. Who is responsible for depleted uranium and low level mixed waste at the Portsmouth plant? Who is responsible for the decontamination and decommissioning costs at the plant.

Response:
The President signed into law on April 26, 1996, Pub. L. No. 104-134, 110 Stat. 1321 (1996). Title III of this Act, namely the "USEC Privatization Act," among other things, deals with the disposal of low-level radioactive waste (LLRW), including depleted uranium, if it were ultimately determined to be low-level radioactive waste, to be generated by facilities regulated by the NRC. According to this Act, DOE is required to take possession of, at the request of the generator, and provide for the ultimate disposition of all such material.

DOE is responsible for decontamination and decommissioning costs once the plant is returned to them.

12.B. Will USEC have to pay the disposal costs for its depleted uranium and other waste?

Response:
According to the Privatization Act, USEC is required to reimburse DOE for disposal of low level radioactive waste (LLRW) and depleted UF_6 (DUF6) - if it is determined to be LLRW - in an amount equal to the costs incurred by DOE, including a pro-rata share of any capital costs.

12.C. Who will receive and manage the radioactive and mixed waste?

Response:
See response to 12.A. and 12.B.

12.D. Who pays for decommissioning? It is unfair for the taxpayers to pay.

Response:
DOE will pay for decommissioning. Current law requires DOE to pay any costs of decontamination and decommissioning (D&D) with respect to conditions existing before July 1, 1993. D&D of any supplemental contamination resulting from accidental spills, leaks, releases, etc. after July 1, 1993, would be the responsibility of USEC.

12.E. What is the anticipated market for enriched uranium?

Response:
NRC's mission does not require it to maintain current information on the anticipated market for enriched uranium, and that question is not part of NRC's safety, safeguards or security evaluations for the gaseous diffusion plants.

12.F. Two resident inspectors at the plant does not seem adequate.

Response:
See response to 6.D.

12.G. Are NRC's health and safety standards available to the public?

Response:
Yes. NRC's health and safety standards are contained in Chapter I of Title 10 of the Code of Federal Regulations (CFR). Specifically for the two operating GDPs, NRC's health and safety standards are contained in 10 CFR Part 76.

12.H. A copy of the USEC proposed compliance plan should have been distributed at the public meeting.

Response:
Due to the size of the Compliance Plan, it was not distributed at the public meetings. It is publicly available at the Public Document Room (PDR), 2120 L Street, N.W., Washington, D.C., and the Local Public Document Room (LPDR), Portsmouth Public Library, 1220 Gallia Street, Portsmouth, Ohio. The final versions of the Compliance Plan are also available at these locations.

12.I. How will NRC regulate AVLIS?

Response: See response to 11.G.

Commenter: Ms. Vina Colley

12.J. Is there plutonium at the Portsmouth plant? Are there fluoride gas releases at the Portsmouth plant?

Response:
Plutonium is present in small amounts as contamination (from past processing of recycled uranium) and in laboratory sources.

There are fluoride gas releases at the Portsmouth plant. First, there are planned releases of fluoride gases up any of 16 stacks under carefully measured and controlled conditions. These releases are regulated by the EPA. Additionally, there are unplanned fluoride gas releases from time to time as a consequence of accidents, human error, or component failure.

12.K. All classified, secret, and proprietary documents associated with the plant should be released to the public.

Response:
Information may be classified and specifically exempted from disclosure by statute (e.g., the Atomic Energy Act of 1954, as amended) or authorized under criteria established by an Executive Order to be kept secret in the interest of national defense or foreign policy.

Section 2.790 (d) of the Code of Federal Regulations provides that correspondence and reports to or from the NRC which contain information or records concerning a licensee's or applicant's physical protection or material control and accounting program for special nuclear material, not otherwise designated as Safeguards Information or classified as National Security Information or Restricted Data, may be withheld from public disclosure.

12.L. Real time monitors should be set up at the site boundaries to detect chemicals and radiation.

Response:
The facility has continuous vent monitors for 13 discharge points which are considered to be the potentially significant contributors to the total plant radionuclide emissions. The plant also maintains both onsite and offsite stations to collect ambient air samples continuously. Samples are collected and analyzed monthly. Thermoluminescent dosimeters are also located both onsite and offsite to monitor external gamma radiation. This program is considered to be adequate.

12.M. Exposures to workers should be as low as reasonably achievable (ALARA).

Response:
USEC is required by 10 CFR Part 20 to maintain radiation exposures to workers ALARA. In addition, USEC has committed in SAR Section 5.3 to do so.

12.N. NRC should have civil penalty authority.

Response. See response to 6.D.

12.O. NRC should have more than 2 inspectors onsite.

Response: See response to 6.D.

12.P. The emergency plan should adequately reflect the presence of nearby schools and institutions.

Response: The Emergency Plan includes U.S. Geological Survey maps that identify the locations of schools and churches.

12.Q. The plant has uranium leaks in its containment.

Response:
A large portion of the enrichment process is operated below atmospheric pressure so that loss of containment would result in inleakage of air into the cascade as opposed to outleakage of UF_6 gas. Significant inleakage, which could cause severe operational upsets, can be detected in many ways before it impacts safety. Where UF_6 exists above atmospheric pressure, sensitive smoke detectors are required to be installed and operable at all times, so that any release of significant quantities of UF_6 would readily be detected. Also the Radiation Protection program requires periodic surveys to detect contamination.

12.R. The public is concerned about safety due to past problems at the plant. NRC should assure that workers and the public are protected.

Response:
The NRC's mission is to assure that USEC provides adequate protection of the workers and the public health and safety and NRC will carry out that responsibility.

12.S. Fluorine used at the plant is extremely hazardous.

Response:
The consequences of a fluorine release have been analyzed and the results reported in the application SAR. No effects are expected to people off site. The fluorine concentrations that would result in the immediate location of a release do have the potential to result in a fatality if a worker is unable to escape. Therefore, workers are trained in the proper operation of the fluorine system and are required to operate the system in accordance with those procedures.

12.T. The method of evacuation if there was an accident is inadequate because school buses might be used.

Response:
The Emergency Plan does not rely on school buses to evacuate personnel from the plant area.

12.U. The plant possesses neptunium.

Response:
Neptunium is one of the many transuranic radioactive substances that the NRC staff has authorized USEC to possess in the form of sealed (0.5 Ci) and unsealed (1.0 Ci) sources and as contaminants resulting from previous operations and projected to result from processing uranium from the Former Soviet Union. The provisions of 10 CFR Part 20 require USEC to handle all radioactive substances in a manner that assures public, worker and environmental protection.

12.V. NRC needs to review the waste treatment process involving thermal absorption.

Response:
The waste treatment process that USEC is using es not include a thermal absorption process.

12.W. Will NRC regulate AVLIS?

Response: See response to 11.G.

12.X. The plant operations involve both chemical and radiological hazards.

Response:
USEC has both radiation safety and chemical safety programs. The NRC staff has determined that USEC's proposed radiation protection program will ensure worker protection from all onsite radiological hazards as well as the toxicological hazards of uranium. NRC will also coordinate with OSHA to assure protection from all chemical hazards.

12.Y. How does NRC plan to measure neutron exposures to workers?

Response:
USEC must measure significant neutron doses to workers as required by 10 CFR Part 20. 10 CFR 20.1502 requires all NRC licensees to monitor personnel for radiation exposures if they are likely to receive in one year more than 10 percent of the annual dose limit. For the whole body, the annual dose limit is 5 rem.

The doses that workers are likely to receive from neutron radiation onsite are well below the monitoring criteria provided in 10 CFR 20.1502. For instance, in the tails cylinder storage areas, USEC reports neutron dose rates between 0.1 and 0.3 millirem/hr. This

dose rate is a small fraction (less than 10 percent) of the dose rates from bremsstrahlung and other direct radiations emitted from full tails cylinders. Also, past history of operations at the GDPs has not indicated a significant neutron dose hazard. See CER Chapter 7.

12.Z. Workers should have a daily urinalysis by an independent monitor.

Response:
USEC has proposed an adequate bioassay program designed to detect soluble uranium intakes below the weekly regulatory limit of 10 milligrams. This conclusion is based on the determination that the lower limits of detections for urine samples, in conjunction with the sample collection frequencies proposed by USEC, are adequate to ensure compliance with the weekly 10 milligram soluble uranium intake limit. Also, past history of operations at the GDPs has not indicated a significant internal dose hazard.

12.AA. The public is concerned about liquid effluents from the plant.

Response:
It is not clear what the nature of the commenter's concern is. Discharges from the plant are made in accordance with the state issued discharge permit. USEC collects water samples as part of its monitoring program.

Commenter: Mr. Tim Mitchell

12.BB. Workers should not be wrongfully disciplined for contamination incidents.

Response:
The NRC's primary responsibility at the gaseous diffusion plants is to ensure that workers and the public are protected from unnecessary or excessive exposure to radiation and that the facilities are operated in a safe manner. The NRC does this by establishing requirements in Title 10 of the Code of Federal Regulations and in the certificate issued to USEC. In general, the NRC does not regulate or otherwise have a role in the hiring and firing practices at the gaseous diffusion plants or other facilities it regulates, with the following important exception. Federal law prohibits an employer from firing or otherwise discriminating against an employee for bringing safety concerns to the attention of the employer or the NRC. Specifically a worker may not be fired or otherwise discriminated against because the worker (1) asks the NRC to enforce its rules against the employer; (2) refuses to engage in activities which violate NRC requirements; (3) provides or plans to provide information to the NRC or the employer about violations of requirements or safety concerns; or (4) asks for, testifies in, helps in, or takes part in an NRC or Congressional or Federal or State proceeding.

Commenter: Mr. Gerald Wilkins

12.CC. Will NRC certify the plant if there is no labor agreement?

Response:
NRC regulations do not require a labor agreement to be in place for the certification.

13. Paducah Public Meeting Transcript - December 5, 1995

Commenter: Ms. Jotilley Dortch

13.A. Will there be a commercial low-level waste storage disposal facility at Paducah?

Response:
No, there will not be a USEC commercial waste facility at Paducah.

13.B. USEC should not be permitted to withhold the waste storage plan from public disclosure.

Response:
See response to 3.H.

13.C. There is a seismic risk at the Paducah plant.

Response:
See response to 2.B.

13.D. An accident could adversely affect the nearby inland waterway.

Response:
It is possible for contamination to occur offsite as a result of an accident, but the risk is very small. However, most of the contamination is likely to occur near the vicinity of the release. USEC would be required to assess any contamination that occurred as a result of an accident and decontaminate the area as appropriate.

Commenter: Mr. Ronald Lamb

13.E. NRC should have civil penalty authority.

Response:
See response to 6.D.

13.F. Does NRC have adequate inspection coverage?

Response:
See response to 6.D.

13.G. Information on waste should be available to the public.

Response:
See response to 3.H.

Commenter: Mr. A.B. Puckett

13.H. Is the NRC an independent regulator, separate from DOE? Is the NRC going to be responsible for the public health and safety and assuring protection of the people near the plant?

Response:
Yes, NRC is independent from DOE. NRC is responsible for assuring that USEC provides protection for people living near the plant. See also response to 12.R.

Commenter: Mr. Mark Donham

13.I. In the past, DOE has not adequately controlled disposal of radioactive and hazardous waste.

Response:
USEC has a radioactive waste management program for safe handling of waste. The NRC will assure that USEC handles its waste safely.

13.J. Public notification about the meeting was not sufficient. People who have expressed interest over the years should have received a notice in the mail.

Response:
The meeting was announced in press releases, Federal Register Notices, and ads in the local paper. In addition, those individuals who have expressed interest in the NRC activities at the site did receive a copy of the Federal Register Notice announcing the meeting.

13.K. The people have a right to know the extent of contamination around the plant site.

Response:
The application contains environmental monitoring data. In addition, DOE publishes an annual Environmental Report that summarizes environmental monitoring data.

13.L. Does the C-310 stack routinely vent radioactive gases?

Response:
The Paducah plant operations result in gaseous effluent from several sources, including the C-310 stack. Emissions are within the regulatory limits. The highest dose to an offsite individual in 1994 was 0.016 mrem, compared to the 10 mrem limit.

13.M. Is there technetium in the groundwater and is neptunium the transuranic waste at the Paducah plant?

Response:
Yes, there is technetium contamination in the groundwater. Both the technetium and the transuranic wastes resulted from processing recycled uranium that contained these contaminants. Recycled uranium is no longer processed at the Paducah plant.

13.N. Are rail shipments coming from Portsmouth and Oak Ridge?

Response:
The Paducah plant ships most of its product to the Portsmouth plant. Portsmouth ships some of its uranium tails back to the Paducah plant for further removal of the uranium-235. The Paducah plant does not routinely ship to or receive items from Oak Ridge. The Oak Ridge facility is shut down and may on occasion ship equipment to the Paducah plant.

13.O. What type of warning system will the NRC require for the public?

Response:
It should be noted that USEC is not required by Part 76.91, "Emergency planning," to have a public warning system and that the inclusion of a public warning system in the emergency plan for the Paducah Gaseous Plant was a voluntary decision made by USEC. The existing public warning sirens do not provide total coverage of the immediate notification area, and plans are in place to replace the existing sirens with a new siren system to ensure better coverage of the immediate notification area. The scheduled completion date for the new system to be installed and operational is March 15, 1997.

13.P. If NRC cannot find the plant in compliance, could it be turned back over to DOE?

Response:
DOE has agreed to retain regulatory oversight over the gaseous diffusion plants until the initial certification is completed, and for an additional transition period to allow for an orderly transition to NRC regulation. After NRC assumes regulatory jurisdiction, regulatory oversight will remain an NRC responsibility until enrichment operations are terminated and the plants are returned to DOE for decommissioning. After initial certification, the plants must retain a valid certificate of compliance and/or an approved compliance plan to continue to operate.

13.Q. NRC should do offsite background monitoring to assess any accumulation of radioactive contamination.

Response:
NRC does not routinely conduct environmental sampling around similarly licensed facilities and does not plan to conduct environmental sampling around the Paducah plant. As part of the inspection program, samples are occasionally analyzed to verify the applicant's sampling and analysis program.

Appendix B INTERAGENCY CONSULTATION RESULTS

Interagency Consultation Results

The Energy Policy Act of 1992 required the NRC to consult with EPA. NRC added to this requirement in 10 CFR §76.53, which requires the NRC to consult with EPA and solicit EPA's written comments on the application. As part of the consultation process, NRC staff met with EPA Region on February 28, 1995 at their offices in Chicago, Ill. Region 5 is the region with responsibility for the Portsmouth facility. Prior to receipt of the application, the staff met with EPA headquarters staff on August 24, 1995; the regional offices (R4 and R5) participated by a conference line. EPA (R5) was provided a copy of the application by letter dated September 15, 1995. The letter invited EPA to comment on the application, informed them of the public meeting, and offered to meet to discuss any questions. In order to inform EPA of changes in the application, by letter dated October 27, 1995, the staff provided copies of USEC responses to questions on the environmental and 4 waste management areas. By letter dated November 7, 1995, a copy of the Compliance Plan was provided to EPA with a request for comment and an offer to meet at EPA's request. EPA responded by letter dated February 29, 1996. EPA raised no substantive objection to either the application or the Compliance Plan.

Although not required by law, NRC staff also consulted with OSHA. OSHA was invited to comment on the application and Compliance Plan by letters dated September 15, 1995, and November 8, 1995, respectively. OSHA provided comments by letters dated November 24, 1995; the response to those comments is provided in Appendix A.

The 1996 USEC Privatization Act requires NRC and OSHA to enter into a memorandum of agreement, within 90 days of enactment of the Privatization Act, to govern the exercise of their authority over occupational safety and health at the GDPs. The staff held several meetings/discussions with OSHA officials in developing a Memorandum of Understanding (MOU); an MOU had been planned prior to the Privatization Act. The MOU describes the authorities of NRC and OSHA in implementing the Act and covers such topics as inspection, investigation, and enforcement. The MOU was signed on July 26, 1996, and published in the Federal Register on August 1, 1996.

The staff also offered to meet with state and local officials and held a meeting for this purpose on November 28, 1995. No state and local official expressed any concern outside the issues already under consideration by the staff.