5.2 NUCLEAR CRITICALITY SAFETY

5.2.1 Introduction

USEC, as part of its application for a certificate of compliance, is required by 10 CFR 76.35(a)(7) to provide a "description of the management controls and oversight program to ensure that activities directly relevant to nuclear safety. . . are conducted in an appropriately controlled manner that ensures protection of employee and public health and safety. . ." In addition, 10 CFR 76.89 requires USEC to maintain and operate a criticality monitoring and audible alarm system. This section describes the basic criticality prevention program and briefly discusses the monitoring provisions at PORTS. Section 3.6.2 discusses the Criticality Accident Alarm System (CAAS) in detail.

5.2.2 Program Elements

The nuclear criticality safety (NCS) program as defined in this section is implemented by the plant NCS procedures. The plant NCS procedures address plant personnel NCS responsibilities, adherence to nuclear criticality safety approval (NCSA) requirements, review and approval of fissile material operations, posting and labeling requirements, response to NCSA violations and NCS training requirements.

5.2.2.1 Adherence With ANSI/ANS Standards

The nuclear criticality safety program has been developed to comply with ANSI/ANS-8.1-1983, ANSI-N16.5/ANS-8.7-1975, and ANSI/ANS-8.19-1984. The issues which are not in full compliance with these standards are addressed in Section 5.2.4.

5.2.2.2 Nuclear Criticality Safety Responsibilities

The General Manager has overall responsibility for NCS and approves the implementation of Nuclear Criticality Safety Approvals (NCSAs). The General Manager assigns responsibilities and delegates commensurate authority to all levels of management for the implementation and oversight of the NCS program.

The organization managers are responsible for ensuring that operations involving uranium enriched to 1 wt. percent or higher ^{235}U and 15 g or more of ^{235}U are identified and evaluated for nuclear criticality safety prior to initiation of the operation. The organization managers or their designees are also responsible for ensuring NCS approvals are requested, and for ensuring 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 approved NCSAs either through pre-job briefings, required reading, training, and/or procedures (based on the complexity of the change). These 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. First-line management is also responsible for removing fissile material handlers (i.e., personnel directly

involved in operations involving uranium enriched to 1 wt. percent or higher ^{235}U and 15 g or more of ^{235}U), who fail the test associated with the NCS training, from jobs which involve handling of fissile material.

Managers are trained in NCS and ensure all appropriate personnel (i.e., fissile material handlers) receive NCS training as specified in the plant NCS procedures. This training provides personnel with the knowledge necessary to fulfill their NCS responsibilities. Section 6.6 discusses the training program in more detail.

The fissile material operators are responsible for conducting operations in a safe manner in compliance with procedures and are required to stop operations if unsafe conditions exist. Stop work and restart authority is discussed in more detail in Section 6.1.

The NCS manager 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 staff members report to the NCS Manager who reports to the USEC NCS Manager. The USEC NCS Manager reports directly to the Director, Engineering. Nuclear Criticality Safety is independent of organizations which require NCSAs. Section 6.1 gives more details related to plant organization.

The specific qualifications for the USEC NCS Manager are discussed in Section 6.1.

Qualified NCS Engineers and Senior NCS Engineers are responsible for performing the following functions: 1) providing NCSAs for fissile material operations, 2) performing facility walk-throughs of facilities which handle fissile material and advising appropriate management 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 Plant Operations Review Committee (PORC), and 6) participating in the review of procedures which involve fissile material operations to ensure NCSA commitments have been effectively incorporated into operating procedures. As does any employee, the Nuclear Criticality Safety group 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 NCS evaluation and approval procedure. These responsibilities are: 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.

5.2.2.3 Process Evaluation and Approval

Each operation involving uranium enriched to 1 wt. 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 a 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 operation completes a NCSA request form (Request for Criticality Safety Evaluation). This documents the operating organization's request for NCS evaluation of the operation. This request is approved and signed by the manager of the operating group or his/her designee. The request is then submitted to Criticality Safety for analysis.

In response to the request, a NCSE is prepared to document the analyses performed as specified in the NCS evaluation and approval procedure. A NCS parameter checklist is used to identify and document potential upset conditions, or contingencies, presenting NCS concerns. Engineering judgement of the qualified NCS engineer may indicate the need for a more detailed study. For example, a HAZOP study may be used if the operation is complex and involves multiple interacting systems which requires substantial input from operations, maintenance, and other subject matter experts to identify the possible upset conditions. A contingency analysis is performed, in which the subcriticality of a process, given the occurrence of the contingency is assessed. This analysis demonstrates the double contingency principle for the proposed operation.

The double contingency principle as stated in ANSI/ANS-8.1-1983, 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 criticality accident is possible." The PORTS NCS program meets the double contingency principle by implementing at least one control on each of two different parameters or implementing at least two controls on one parameter. Controls include passive barriers (e.g., structures, vessels, piping, etc.); active engineered features (e.g., valves, thermocouples, flow meters, etc.); reliance on the natural or credible course of events (e.g. by relying on the nature of a process to keep the density of uranium oxide less than a specified fraction of theoretical); and administrative controls that require human actions to be taken in accordance with approved procedures, or by other means that limit parameters within specified values. If 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.

The basis for a parameter or process condition change that could lead to a criticality being unlikely must be documented in the NCSE. The basis may be an engineered feature, administrative control, the natural or credible course of events, or any combination of these or other means necessary to ensure the change is unlikely to occur. The parameters or conditions relied on and the limits must be specified in the NCSE and controlled.

Where the natural or credible course of events is relied upon in whole or in part to prevent a process condition change, the factors that influence the process must be described in sufficient detail in the NCSE as items related to NCS and programmatically controlled. For items which are established, maintained and implemented by non-NCS programs, credit for availability and reliability is established as described in SAR Section 6.3.5, Physical Plant Change Control and Configuration Management, without the necessity of establishment of additional NCS controls. For situations where the NCS-credited controls do not provide adequate assurance of availability or reliability (i.e., situations where non-NCS programmatic and physical plant changes could adversely affect the intended criticality safety function of the items relied upon for criticality safety), specific NCS controls will be established, maintained and implemented to ensure criticality safety.

The NCS evaluation is performed by a qualified NCS engineer. Qualified NCS Engineers meet the requirements specified in the NCS qualification procedure. The minimum requirements for a qualified NCS Engineer are: 1) baccalaureate in engineering, mathematics, or related science; 2) familiarization with the facility by completing a minimum of 1 year in a GDP Criticality Safety section; 3) completion of NCS-related training course and KENO V.a training course or equivalent; 4) performance of at least four evaluations under the direction of a Senior NCS Engineer; 5) performance of walk- through inspections under the guidance of a qualified NCS engineer, and 6) 1 year of organized training in the physics of nuclear criticality and in associated nuclear safety practices if the trainee does not have a nuclear engineering or physics background. The Criticality Safety Manager can modify the minimum qualified NCS Engineer qualification requirements for personnel who have worked for a minimum of 3 consecutive years at other facilities as a nuclear criticality safety engineer.

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

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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.

Engineering judgment of both the analyst and the technical reviewer is used to ascertain independence of events and their likelihood or credibility. The basis for this judgment 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.

Once the NCSE is completed, a technical review of the evaluation is performed and documented. The Criticality Safety 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 or is a NCS engineer completing the technical review under the guidance of a Senior NCS Engineer. The minimum requirements for a qualified Senior NCS engineer are: 1) completion of the minimum requirements for a qualified NCS engineer; 2) performance of the functions of a qualified NCS engineer; 3) completion of 1 year as a qualified NCS engineer and 2 years' gaseous diffusion plant experience; and 4) be approved by the Criticality Safety Manager. The Criticality Safety Manager may modify the minimum Senior NCS Engineer qualification requirements for personnel who have worked for a minimum of 5 consecutive years at other facilities as a nuclear criticality safety engineer.

The NCSA is prepared based on the results of the NCSE, and it 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 in accordance with TSR Section 3.11.4, regarding double contingency. 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 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. Editorial changes require only the approval of the NCS Manager. Editorial changes are defined as changes which do not change the technical basis of the NCSE/A. The PORC reviews the NCSA to ensure consistency with other NCSAs and other potentially conflicting requirements or regulations. 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 Criticality Safety personnel. The documentation of this verification process is maintained as a quality record along with the NCSE. The manager or designee of the operating group then signs the NCSA, indicating acknowledgment and agreement with the limits and controls specified. The NCSA is then issued as a permanent or temporary document.

First-line management of the operating organization is responsible for implementing, through training and procedures, the conditions delineated in the NCSA. Operational aids such as postings, labels, boundaries for fissile material operations, and fissile material movement guidelines are provided as specified in the NCSA. First-line management ensures postings and labels are prepared and verifies

that they are properly installed as required by the NCSA. The procedures are prepared or modified to incorporate the NCSA requirements. First-line management is responsible for ensuring the employees understand the procedures and understand the NCS requirements before the work begins.

Each completed NCSA is issued as a controlled document. The NCSAs are maintained in a controlled manual which is issued to the responsible facility manager. Completed NCSEs and NCSAs are archived and retrievable as permanent quality records in accordance with the plant records management system described in Section 6.10. The NCSA/NCSE process provides assurance that operations will remain subcritical under both normal and credible abnormal conditions. A summary of the NCS controls and parameters controlled as well as Active Engineeered Functions based on approved NCSAs/NCSEs are presented in Appendix A.

There are three operations which do not meet the double contingency principle as described earlier in this section. However, these operations have been evaluated to be safe, and are discussed in the following paragraphs. These operations are: the handling, storage, and transportation of large product cylinders; operation of the cascade equipment; and handling of large cascade equipment items (e.g., compressor, converter, G-17 valve, etc.) which have large deposits of uranium. Summaries of the accident scenarios and NCS controls associated with operation of the cascade equipment, handling of large cascade equipment items, and handling, sotrage and transportation of UF_6 product cylinders are addressed in Appendix A.

The safety of the product cylinders is based on moderation control provided by the UF_6 withdrawal process and maintained by ensuring cylinder integrity. Operation of withdrawal station compressors at pressures that do not exceed 45 psig guarantees moderation control that is verified by measurement. The H/U ratio will be maintained at 0.088. Any decrease in temperature at this pressure would result in solidification of UF_6 and termination of the withdrawal process. Various Technical Safety Requirement (TSR) controls are in place to ensure cylinder, thus precluding the introduction of water, including: Lmiting Conditions for Operation for systems controlling cylinder heating temperature and cylinder pressure; Design Features and asociated Surveillance Requirements for cylinders that are to be filled, liquid cylinder cranes and liquid cylinder lifting fixtures; and Administrative Controls for cylinder fill wieghts and cylinder handling. Additionally, cylinders are inspected for integrity, in accorance with established procedures. Additionally, TSRs are in place to control moderation in cascade equipment with ^{235}U mass greater than safe mass during shutdown conditions and in removed cascade equipment with ^{235}U mass greater than safe mass.

Both the operation and removal of cascade equipment with a uranium deposit greater than a minimum critical mass (mass required for criticality with optimum moderation, reflection, and geometry) could go critical if sufficient moderator (from any moderation source) should enter the equipment. While optimum geometry and reflection are not anticipated, it is not practical to exercise direct control over these parameters.

There are three sources of moderation that are of different levels of concern: moisture contained in air that leaks into the cascade, oil inleakage from the lubrication and hydraulic systems and water inleakage from the RCW system.

A dry environment is provided by UF_6 for operating equipment within the cascade and by treatment gases in equipment undergoing treatment. If wet air should enter the equipment, the amount of hydrogen,

generated by the reaction with UF_6, that is retained in the vicinity of the deposit is too small to be a hazard. In the case of an operating cell, HF formed by the reaction of moisture with UF_6 is quickly removed from the equipment by the separative capability of the cascade. Equipment which contains greater than a safe mass of U^{235} and is not operating is filled with a dry gas to minimize wet air inleakage. In the case of a cell treatment, the duration of the treatment coupled with the leak rate criteria limit the amount of HF present.

Cascade equipment is maintained to prevent oil inleakage to the cascade. Oil lines are routed around equipment to ensure that any leak that would develop in a line would leak to atmosphere rather than into the process. As a result of equipment failures, oil has leaked into the process from failed bearings and seals on compressors. However, these types of catastrophic failures are quickly isolated and properly handled. The risk associated with oil inleakage in equipment is less in non-operating equipment than in operating equipment.

Process parameters are monitored to identify process gas deposition in a timely fashion to prevent the formation of significant deposits. In addition, monitoring programs are in place to routinely measure radiation levels at locations in the cascade that are likely to develop deposits. These measurements are taken monthly in the X-326 and X-330 buildings and quarterly in the X-333 building. These programs are the responsibility of Operations.

Water from the recirculating cooling water (RCW) system is protected from entering a cell with R-114 since the pressure of the R-114 coolant is maintained higher than process gas (UF_6) pressure and the RCW pressure. To have a criticality occur from inleakage of RCW, it would be necessary for the coolant pressure to drop below the RCW pressure, remain above the process gas pressure and leaks to develop in both systems. Additionally, the operator would have to fail to respond to a low pressure coolant alarm or compressor surging. Therefore, double contingency is provided for the case of a criticality by entrance of water via the coolant system.

Similar types of controls are maintained over the RCW for non-operating equipment. In some cases, they are the same as those described for an operating cell. In cases when the R-114 must be drained from the coolant system, the coolant condenser is drained to remove the risk of a criticality. As for the operating cell, double contingency is provided for the case of a criticality by entrance of water via the coolant system in the non-operating cell. However, a low pressure coolant alarm and compressor surging may not be available as indicators of RCW inleakage.

When equipment is removed from the cascade, a dry gas buffer or dry air purge is maintained in the equipment until equipment disassembly can begin. During disassembly, exposed uranium-bearing deposits are removed into favorable geometry containers. This minimizes the time of exposure of the uranium-bearing materials to ambient air conditions.

New operations and operations other than those identified above as not meeting the double contingency principle shall comply with the double contingency principle.

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 orally or in writing. Such direction can include a stop work order or other appropriate instructions. Documentation will be prepared within 48 hours after the emergency condition has been stabilized.

5.2.2.4 Design Philosophy and Review

Designs of new fissile material equipment and processes must be approved by Nuclear Criticality Safety 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 use 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. The adherence to this approach is determined during the preparation, and technical review of the NCSE performed to support the equipment design. This preferred design approach is implemented as described in plant NCS procedures.

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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 to satisfy the double contingency principle for those cases where the preferred design approach cannot be met.

5.2.2.5 Criticality Accident Alarm System Coverage and Nuclear Accident Dosimetry

A 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 the total personnel exposure to emitted radiation over the course of the accident.

At PORTS, the CAAS detects neutron dose rate. 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. These instruments are described in Section 3.6.2.

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, 50 g of ^{235}U in any square meter of floor or ground area, 5 g of ^{235}U in any 10-liter volume, or areas having material that is either packaged and stored in compliance with 10 CFR 71 or specifically exempt according to 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 wt. percent or higher ^{235}U and 15 g or more of ^{235}U do not require a NCSE and are not required to have CAAS coverage.

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 detectors at 5 millirad per hour above the background radiation rate for the area(s) of coverage.

The location of detectors and setpoints is based on results of dose calculations and detector tests performed at critical experiment facilities. These tests and calculations demonstrate that the CAAS will respond in accordance with the detection requirements specified in 10 CFR 76.89.

Additional details describing the criticality accident alarm system equipment, operation, testing, maintenance, and locations are provided in Section 3.6.2. Reference Section 5.3 for a discussion of the nuclear accident dosimeters.

5.2.2.6 Procedure Requirements

Operations to which NCS pertains shall be governed by written procedures or job task checklists. These procedures or checklists 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 procedure. NCSA requirements are identified by placing a "commitment stamp" in the left hand margin next to the requirement. Identifying these requirements in this way ensures changes to these requirements are not made without review and approval by Criticality Safety. The NCSA requirements are incorporated into the appropriate procedures or job task checklists as required by the NCS evaluation and approval procedure.

New and modified procedures are reviewed by the appropriate safety organizations, including Nuclear Criticality Safety, as specified in the procedure for procedure control. Criticality Safety 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 recommends approval of a procedure prior to issuance as specified in SAR Section 6.11.4.5 and TSR Sections 3.9.1, 3.9.2 and 3.10. Reference Section 6.11 for more details related to the procedure development and change control program and Section 6.2 for the PORC description.

5.2.2.7 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 procedures. Approved NCSAs specify the wording for the postings. Labels are prepared in accordance with the plant NCS procedures and used as required by NCSAs. Limits and controls are printed or written in an appropriate size, and the postings and labels are placed in conspicuous locations determined by the line organization.

5.2.2.8 Change Control

A Configuration Management (CM) Program ensures that any change from an approved plant baseline configuration is managed so as to preclude inadvertent degradation of safety or safeguards. The CM Program, described in detail in Section 6.3, includes organizations and administrative processes to ensure accurate, current design documentation that matches the plant's physical configuration while complying with applicable requirements. The CM Program applies to NCS, both because a change to a system, structure, or component (SSC) controlled by CM may require a NCS approval, and because the documents generated by the NCS Program—NCSAs and NCSEs—become controlled documents, themselves subject to CM 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 manager, etc.) planning the change to determine if the change affects structures, systems, and components (SSCs) relied on for safety. The design control process specifies the organizations required to perform reviews of changes to SSCs. The required approvals are obtained before the change is implemented. Design engineering management verifies the required reviews have been performed before approval. If an item is relied on for the criticality safety of an operation, as a barrier for either double or single contingency, it will be identified through the work control process as an NCS SSC, and Criticality Safety approval is required before implementing the change. Criticality Safety reviews the NCSE for this specific operation and determines if the change affects the analysis performed and the conclusions made in the NCSE. The change request will be approved by Criticality Safety 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 systems which require configuration change control are identified as Q or AQ-NCS. The configuration change control program is discussed in more detail in Section 6.3.

Document control is another element of the CM Program. Procedures, documents, and records control programs provide for centralized control and issuance of documents critical to the maintenance of the design history, and a repository for records to verify this maintenance. NCSAs are specifically included in the index of documents that are required to be controlled, which is maintained by Plant Services.

5.2.2.9 Operation Surveillance and Assessment

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

NCS walk-throughs of facilities that may contain fissile material operations are performed by Criticality Safety personnel to determine the adequacy of implementation of NCS requirements and to verify that conditions have not altered to adversely affect NCS. These walk-throughs are performed as specified by the Criticality Safety procedure on walk-throughs. 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 Criticality Safety personnel. As a minimum, these walk-throughs are completed for applicable areas biennially and may be performed in conjunction with the assessments discussed below.

At a minimum, fissile material operations are reviewed for NCS on an annual basis. These reviews are performed as part of a self assessment performed by organizations operating with fissile materials. Assessments involve both the operating organization and Criticality Safety personnel. These assessments include the inspection of facility modifications and changes, operating procedures, and compliance with NCSAs. These assessments are performed as specified by the plant NCS procedures.

Internal audits of the NCS program are conducted or coordinated by the Nuclear Safety and Quality Manager. These audits will be performed at a frequency of at least every three years. The purpose of these audits is to determine the adequacy of the overall NCS program. This includes the adequacy of the NCSEs, NCSAs, internal assessment programs, and implementation of the NCS

requirements.

The results of these walk-throughs, assessments, and audits are documented and reported to appropriate managers. Identified deficiencies are documented and corrected according to the problem reporting system described in Section 6.9 and Section 2.18 of the Quality Assurance Plan.

NCS deficiencies are recorded and the data trended to monitor and prevent future violations. Deficiencies are grouped into categories such as organization, building, type of process, and type of deficiency. Corrective actions will be taken for adverse trends in accordance with the Quality Assurance Plan.

PORC provides review of the NCS program in accordance with the Technical Safety Requirements.

5.2.3 Technical Aspects

5.2.3.1 Application of Parameters

Moderation

Water and oil are considered to be the most efficient moderators commonly found on the plantsite. When moderation is not controlled, either optimum moderation or worst credible moderation is assumed as the normal case when performing analyses. When moderation is controlled, credible abnormal process upset conditions shall determine the worst case moderated conditions.

Moderation control is applied to enrichment cascade equipment and product cylinders.

The enrichment cascade is a closed system designed to process gaseous UF_6. This closed system prevents the introduction of moderation due to wet air inleakage. Also, because UF_6 reacts chemically with moisture (a moderator) to produce solid uranium-bearing compounds which impedes the proper operation of the cascade, the entire enrichment cascade is designed to minimize introduction of moisture. This includes the use of Freon-114 for coolant rather than water.

Moderation control is the primary NCS control for product cylinders and is based upon the ability of the gaseous diffusion plants to produce high purity UF_6 (greater than or equal to 99.5 percent). Assuming the remaining 0.5 percent impurity was composed entirely of HF, it would not provide sufficient moderation (i.e. atomic ratio greater than 0.088 H/U) of the neutrons to sustain a nuclear chain reaction. The product withdrawal system pressure is limited to less than 45 psig while UF_6 is being condensed for withdrawal to ensure that an atomic ratio of less than or equal to 0.088 H/U is maintained in product cylinders.

Volume

Where volume is used as a control, corresponding volume limits used at PORTS and ^{235}U enrichment limits are as follows:

Volume Limit	Maximum ^{235}U Enrichment (wt. percent)
55-Gallon	1.4
30-Gallon	2.0
22-Quart	5.5
14-Quart	10.0
5-Quart	not limited

The bases for the container sizes are provided in each NCSE prepared for those operations requiring containers. Specific details of these bases can be obtained by referring to the particular NCSE of concern.

Interaction

Interaction is controlled by spacing items bearing fissile material when those items could result in a criticality accident if not properly spaced. The spacing necessary to maintain a safe array of fissile material units is determined in the NCSE performed for the array. The spacing requirements are documented in the NCSA for the operation. The amount of spacing needed between items is determined based on analysis of the normal and credible abnormal process upset conditions for the particular operation. The basis for the spacing is documented in PORTS NCSEs. Other spacing requirements are applied on a case-by-case basis, depending on the results of a given NCSE.

Geometry

Geometry control is applied by limiting equipment dimensions for those systems which depend on the geometry for criticality safety. The geometry is determined in the NCSE which is performed for each system and depends on the normal and credible abnormal process upsets conditions related to the specific system. For example, the following dimensions are used in geometry controls:

Volume Limit	Maximum ^{235}U Enrichment (wt. percent)
5-Inch Depth	5
3.5-Inch Depth	10
1.5-Inch Depth	not limited
10.25-Inch Inside Diameter (ID)	5
8.2-Inch ID	10
5-Inch ID	not limited

The inside diameters are for geometries of unspecified length. Containers with larger inside diameters and limited heights are used as evaluated in NCSEs. Geometry controls are specified in the NCSAs.

Mass

Mass controls are applied on a case-by-case basis depending on the fissile material operation involved. The acceptable mass is determined based on the specific NCSE performed for the operation. The safe mass value depends on many factors including the geometry, the ^{235}U enrichment, composition, etc. The safe mass values are communicated to the operating personnel via the NCSAs.

A typical operating limit is 350 g ^{235}U, regardless of enrichment. A maximum mass of 760 grams ^{235}U is considered subcritical, as recognized by ANSI/ANS-8.1. If under an abnormal condition the mass would not exceed 760 grams ^{235}U, the operation would be considered to be subcritical.

Enrichment

Uranium-containing material on the plantsite with ^{235}U enrichment ≤ 1 wt. percent is considered incapable of supporting a nuclear chain reaction, but interaction of such materials with materials of higher enrichment is taken into consideration in the specific NCSE for those operations which involve material enriched to ≤ 1 wt. percent.

At PORTS, the maximum ^{235}U enrichment of product UF_6 is 10.0 wt. percent. Small quantities of higher enriched uranium may be present on the plantsite in the form of laboratory samples and standards for instrument calibration purposes and in Department of Energy (DOE) operations as described below.

DOE operations involving greater than 10.0 wt. percent ^{235}U on plantsite are ongoing. For NCS, at interfaces between uranium enriched to greater than 10.0 wt. percent ^{235}U and uranium of less than 10.0 wt. percent ^{235}U, either specific controls are present to limit enrichment to less than 10.0 wt. percent ^{235}U, or the possibility of the higher enrichment is addressed in the NCS evaluation for the operation. When 5", 8", and 12" cylinders with residual UF_6 of greater than 10.0 wt. percent ^{235}U are cleaned in X-705, the HEU solutions generated are mixed with LEU solution and double sampled to confirm the ^{235}U enrichment is less than 10.0 wt. percent. In some cases, equipment removed from X-326 could contain residual deposits of HEU. The HEU deposits individually will not be of NCS concern since the HEU suspension project will reduce HEU deposits to below a minimum critical mass, including measurement error for the enrichment of the deposit. In X-705, this equipment would be disassembled, and the bulk of the deposit removed to favorable geometry containers. The equipment would then be decontaminated further in the tunnel cleaning operation. Small amounts of HEU may have entered the seal exhaust oil and trapping material in X-326 during initial HEU refeed and during HEU suspension activities. Due to the large quantity of LEU present, the enrichment would not exceed 10.0 wt. percent in these materials. The operations involved were originally approved for enrichments of up to 100 wt. percent ^{235}U.

The maximum ^{235}U enrichment for each operation is established by the specific NCSE. The NCSA shall specify the maximum acceptable enrichment for each operation. Credible process upset conditions which could alter the ^{235}U enrichment shall also be considered in the NCSEs. Many operations at PORTS were previously approved for enrichments higher than 10.0 wt. percent ^{235}U. Those operations evaluated assuming this higher enrichment will have a larger margin of safety than indicated in the evaluation, due to the lower enrichment of the uranium actually being processed.

In evaluations of operations where ≤10.0 wt. percent ^{235}U is the normally expected enrichment, but that are subject to higher enrichments under abnormal conditions, the abnormal events are evaluated as a contingency.

Density

The density of materials used in a given operation is justified in the NCSE for the operation being considered. If the density must be controlled to maintain compliance with the double contingency principle, it will be documented in the specific NCSA for the operation.

UF_6 in the gaseous phase, at pressures and temperatures existing in the enrichment cascade equipment, is incapable of supporting a nuclear chain reaction even when intermixed with hydrogenous material (e.g., HF). The basis for this conclusion is documented in the NCSE for the operation of the cascade. UF_6 in the gaseous phase in enrichment cascade equipment has low material density and is not exposed to exotic reflecting materials (e.g., beryllium).

Heterogeneity

Heterogeneous configurations are considered for those operations which involve small fissile material and moderator regions. Heterogeneous groupings may occur for the handling of small sample containers; however, 10 wt. percent ^{235}U would be assumed for these samples that are handled on a safe mass basis. Using the homogeneous safe mass at 10 wt. percent ^{235}U would also be safe for heterogeneous 10 wt. percent ^{235}U since at this enrichment, the homogeneous and heterogeneous minimum critical masses are close in value.

Concentration

Concentration controls are used on a case-by-case basis. When the criticality safety of an operation depends on the concentration of fissile material, the medium is sampled twice, the samples are verified to be properly taken by a second individual, and the two samples are independently analyzed as required by the specific NCSA for the operation involved. The specific controls and details are documented in the NCSA for each operation which relies on concentration controls. A typical operating limit is 5 g ^{235}U per liter, regardless of enrichment. A concentration of 11.6 g ^{235}U per liter is considered subcritical, as recognized by ANSI/ANS-8.1. If under all postulated conditions, the concentration would always be less than 11.6 g ^{235}U per liter, the operation would be considered subcritical.

Reflection

Normal and credible abnormal reflection is considered when performing NCS evaluations. The possibility of full water reflection is considered when performing analyses. It is recognized that concrete can be a more efficient reflector than water, and its potential presence is considered.

Neutron Absorption

When neutron absorbers are used as NCS controls, the intended distributions and concentrations under both normal and credible abnormal conditions are maintained in accordance with the requirements of the applicable NCSA. These requirements are: representative sampling of the neutron absorber,

sampling at a frequency based on the environment to which the neutron absorber is exposed, analyzing of samples for all material attributes for which credit is taken in the NCSE, and periodic inspections of fixed neutron absorbers to ensure adequate distribution as specified in the NCSA. For borosilicate glass raschig rings, the rings are used in environments where they would normally be exposed to water only. Inspections are made of the rings for settling and damage on an annual basis. If a UF_6 release should occur, exposing the rings to a corrosive environment, the rings would be replaced with certified rings, or the rings would be tested and verified acceptable according to the testing requirements of ANSI/ANS-8.5. Before a neutron absorber other than Raschig rings 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 and approval.

A NCS evaluation can take credit for the neutron absorption properties of the materials (1) added specifically for the purpose of absorbing neutrons, and (2) of construction, provided an allowance has been made for manufacturing and dimensional tolerances, corrosion, chemical reactions, and uncertainties in the neutron cross-sections.

5.2.3.2 Methods of Calculation

Experimental Data

Experimental data are not specific enough to allow evaluation of operations performed at PORTS. The generic nature of the experimental data does not address the variables present in the different operations. However, experimental data are used for validation of the computer code (i.e., KENO V.a) which is used to perform the calculations needed to support the development of NCSEs. The experimental data used are discussed in the code validation report.

Handbooks

Handbooks are also used in some cases when simple systems are being evaluated. Full advantage of these handbooks can not be taken since most of the operations performed at PORTS are too complicated to be adequately addressed by data in a handbook. When isolated operations are performed with small amounts of fissile material, referencing handbooks is useful for supporting the conclusions made in the NCSE. Examples of the handbooks used include, but are not limited to, ARH-600 "Criticality Handbook" and LA-10860 "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U."

Hand Calculations

Applicable methods for evaluating single units include Modified Two Group Diffusion Equation (i.e., Critical Equation), Buckling Conversion, and Comparative Analysis.

Modified Two Group Diffusion Equation — This method is applicable to, and most widely used for, solution systems.

Buckling Conversion — The method of buckling conversion or shape conversion is applicable to all materials.

Comparative Analysis — This method involves direct comparison of the system configurations to subcritical data from NCS handbooks.

Applicable methods for evaluating arrays include the Solid Angle Method and the Surface Density Method using unit shape factor.

Solid Angle Method — This method is applicable to solution systems. It is not useful if reflection is more effective than a thick water reflector located at the array boundary.

The conditions that must be satisfied in order to successfully apply the solid angle method are (1) k_{eff} of any unreflected unit does not exceed 0.80, (2) each unit is subcritical when completely reflected by water, (3) the minimum surface-to-surface separation between units is 0.3 m, and (4) the allowed solid angle does not exceed 6 steradians.

Surface Density Method using unit shape factor — This method can be used as an approximation for large arrays of identical units containing solutions and metals. This method determines the spacing and mass of units independent of the number of units. An important feature of the Surface Density Method is that it is equally applicable to more irregular geometries, including those typically associated with process equipment.

Computer Calculations

For those cases where adequate references are not available, NCS computational analyses are performed, which involve the calculation of k_{eff} to determine whether 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.

Validation of the computer code determines its calculational bias or uncertainty as well as the effective margin of subcriticality. The PORTS validation involves the modeling of benchmark critical experiments over a range of applicability. Because the k_{eff} value of a critical experiment is essentially 1.0, the bias of the code is taken to be the deviation of the calculated values of k_{eff} from unity. Statistical analysis is employed to estimate the calculational bias, which includes the uncertainty in the bias and uncertainties due to extensions of the area of applicability, as well as the effective margin of subcriticality. Uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data. The validation of the computer code specifically defines the maximum acceptable k_{eff} used to determine subcriticality.

The margin of subcriticality used at PORTS shall result in a k_{eff} upper safety limit which will ensure that there is a 95 percent confidence that 99.9 percent of all future k_{eff} values less than this value will be subcritical. The minimum margin of subcriticality of 0.02 in k_{eff} shall be used to establish the acceptance criteria (i.e., upper safety limit) for criticality calculations. The upper safety limit for PORTS calculations is established as $\leq$ .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.

The calculation of k_{eff} is accomplished by the use of computer codes which utilize Monte Carlo techniques to determine k_{eff} of a system. Computer models representing the geometrical configuration and material compositions of the system are developed for use within the code. The development of appropriate models based on both normal and credible abnormal process conditions is of primary importance in determining the potential reactivity of the system.

When NCS is based on computer code calculations of k_{eff}, controls and limits are established to ensure that the maximum k_{eff} complies with the applicable code validation for that type of system being evaluated.

Scoping and analysis calculations can be performed utilizing various unvalidated computer codes; however, computer calculations of k_{eff} used as the basis for NCS approvals are confirmed by, or performed using, configuration-controlled codes and cross-section libraries for which documented validations are performed with at least the same degree of conservatism as that presented in the validation report POEF-T-3636, Revision 1, and are in accordance with the American National Standards Institute (ANSI) standard ANSI/ANS-8.1-1983.

The configuration control program used to maintain the computer codes and cross sections used in performing k_{eff} calculations meet 10 CFR 76.68 change requirements and consists of the following elements. The System Administrator, a NCS engineer, is responsible for controlling access to the software. The software configuration control and testing program is implemented through the software configuration control plan for the Nuclear Criticality Safety Software (NCSS). Quarterly, one of the following is performed: a bit-by-bit comparison of the production version of the software (executable modules and data libraries) versus an archived production version; or a bit-by-bit comparison of the output from all validation cases versus archived output of all validation cases from the original validation performed when the production version was installed. All modifications to the NCSS require approval of the Software System Team, which includes the System Administrator.

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5.3 RADIATION PROTECTION

As a condition of certification, 10 CFR 76.60(d) requires that USEC "comply with the applicable provisions of 10 CFR Part 20, "Standards For Protection Against Radiation" not later than the date of the Directors decision on the initial application and/or as specified in an approved plan for achieving compliance. This section describes USEC's Radiation Protection (RP) Program for keeping occupational radiation exposures and radioactive contamination as low as is reasonably achievable.

5.3.1 Radiation Protection Program

5.3.1.1 ALARA Policy

In accordance with 10 CFR 20.1101, USEC has established an RP program designed to protect personnel entering the USEC-leased spaces from unnecessary exposure to ionizing radiation and radioactive materials. Management has adopted the following principles:

1. Personnel radiation exposures and the release of radioactive effluents shall be maintained in accordance with the As Low As Reasonably Achievable (ALARA) principle.

2. No individual shall receive a radiation dose in excess of any regulatory limit.

Responsibility for oversight and adherence to this policy rests with the Vice President, Operations. The General Manager has the overall responsibility and authority for the ALARA program at PORTS. The PORTS Radiation Protection Manager is responsible for implementing the ALARA policy.

5.3.1.2 Radiation Protection Committee

The Radiation Protection Committee (RPC), is an independent advisory group to the General Manager, Transfer and Shipping Plant Manager, and the PORC on radiation protection issues, including the ALARA Program. It functions to (1) monitor selected operational radiation protection issues; (2) advise plant management on radiation protection concerns; (3) review proposed designs, work practices, selected suggestions, and selected projects with regard to contamination control and/or ALARA.

<u>Membership and Structure</u>

Membership consists of persons from various functional disciplines of the plant who have the necessary competence and experience to perform the functions of the committee. Standing committee members are the Radiation Protection Manager (RPM) who serves as the chair, the vice-chair who is appointed by the RPM, the Nuclear Safety Manager, the Production Support Manager, Operations Manager, Maintenance Manager, and the bargaining unit representative(s). Participation from other functional disciplines may vary depending on the issue of concern. The committee chair, or designee, is responsible for requesting appropriate functional representation. Committee members may designate an alternate to attend committee meetings in their place.

Quorum

The quorum consists of five standing committee members or their alternates.

Authorities

Committee authority is limited to reviews and recommendations. The committee has no approval, stop, or start work authority. Ad hoc subcommittees may be established for special studies or reviews pertinent to committee-related issues.

Meeting Frequency

The committee meets at least quarterly and as directed by the chair.

Responsibilities

The RPC chair ensures that functions of the committee and tasks which may be assigned by, but are not limited to, the General Manager, Transfer and Shipping Plant Manager, and PORC are properly executed. Minutes are made available to the General Manager, the Transfer and Shipping Plant Manager, and the PORC. Special reports are prepared upon request of the General Manager, the Transfer and Shipping Plant Manager, or the PORC chair, or when the RPC chair determines issues warrant attention.

The RPC reviews matters that have or may have an impact on contamination control and/or ALARA. These include, but are not limited to, the following: (1) Technologies for selected buildings and/or job tasks; (2) current work practices and completed tasks which have/had contamination control or ALARA concerns; (3) radiation protection violations; (4) lessons learned; (5) trends and resulting impacts on contamination control and/or ALARA; (6) establish annual contamination control and exposure goals; and (7) review BEQs as required by SAR Section 5.1.

Minutes and Records

Minutes are issued that identify RPC members and/or alternates in attendance, agenda items, a summary of decisions made, and action items. Copies are made available to the General Manager, the Transfer and Shipping Plant Manager, the PORC, and the RPC members.

5.3.1.3 Radiation Protection Program Elements

PORTS maintains an RP Program[1] that includes the following elements:

1. Policy statements;
2. ALARA program;
3. Approved, written, and controlled procedures that implement the program;
4. Personnel radiological training, including General Employee Radiological Training (GERT);
5. Surveys and evaluations of radiological and soluble uranium's toxicological conditions;
6. Access controls;
7. Administrative radiation and soluble uranium's toxicological dose control levels with provisions for management review and concurrence;
8. A self assessment function designed to monitor personnel adherence to established procedures and radiological controls;
9. Use of protective clothing and personal protective equipment;
10. Inspections and audits;
11. Personnel radiation and soluble uranium's toxicological exposure monitoring;
12. Recordkeeping; and

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[a] Nonoccupational Radiological Protection monitoring programs as described in Section 5.1.

13. Reports.

In accordance with 10 CFR 20.1101(c), the radiation protection program content and implementation, as described in this Section, is reviewed annually. The RP Manager is responsible for this annual review and preparation of a report documenting the results of the review. The report is then reviewed by the Radiation Protection Committee. Revisions to the RP program as described in this Section, if required, are initiated by the RP Manager and submitted to the PORC as part of the annual review.

5.3.1.4 Radiological Protection Organization

The Health Physics (HP) organization provides radiological protection support to the facility, is independent of the organizations responsible for production or remediation, and has an equivalent reporting level. The HP organization provides oversight and control of the technical aspects of the program elements that affect radiological protection.

The RP Manager is responsible for establishment of the RP Program and has the authority to deny access to radiological areas for personnel who do not adhere to radiological protection requirements. The RP Manager has technical oversight of all radiological protection procedures with the authority to oversee, stop work, or subcontract the services, as necessary, to maintain the integrity of the radiological protection program. The RP Manager provides technical guidance for the PORTS RP activities, and is responsible for the RP program at PORTS for the assurance of regulatory compliance. This individual reports to and is accountable to the Production Support Manager. The RP Manager has direct access to the PORTS General Manager and Transfer and Shipping Plant Manager for radiological control matters. The RP Manager's position is shown in Figure 6.1-1, "Uranium Enrichment Facilities Organizational Chart." Responsibilities of the RP Manager described in this section will be carried out by the RP Manager or his/her designee.

The RP Manager and designee are required to have the technical competence and experience to establish radiological protection programs and the management capability to direct the implementation and maintenance of radiological protection programs. The minimum qualifications for the RP Manager and designee are identified in Section 6.1. Certification by the American Board of Health Physics is desirable, but not required.

Support personnel who provide HP and radiological engineering, dosimetry, bioassay, independent oversight, and instrumentation and calibration functions are required to have technical qualifications pertinent to their assigned duties.

HP Technicians and their managers perform the functions of assisting and guiding workers in the radiological aspects of the job. HP Technicians and their managers are qualified and trained in accordance with an approved qualification and training program (see Section 5.3.1.8).

HP Technicians and their managers have the responsibility and authority to stop radiological work or mitigate the effect of an activity if they suspect that the initiation or continued performance of a job, evolution, or test will result in the violation of approved radiological protection requirements.

5.3.1.5 Radiological Protection Procedures

Approved written procedures are prepared and issued to implement the RP program. These procedures are prepared consistent with the requirements of 10 CFR 20 and are approved, maintained, and adhered to for operations involving personnel radiation exposure and toxicological exposure to soluble uranium. These procedures are prepared, maintained and made available to appropriate personnel at the plant as described in Section 6.11.

5.3.1.6 Radiological Work Permits

Radiological Work Permits (RWPs) are required for work activities in Contamination, High Contamination, Airborne Radioactivity, Radiation, and High Radiation Areas. The RP Manager designates HP personnel who are authorized to approve, issue, update, revise, and close RWPs.

The limits established for contamination control (surface and airborne) are based on the toxicity of uranium. The contamination control program, of which RWPs are a part, is designed to ensure that the inhalation or ingestion of soluble uranium are below limits. Therefore, the RWPs as well as other aspects of the contamination control program are based on the toxicity of uranium.

RWPs serve as a basic implementing tool of the RP program and are a primary mechanism by which radiological controls are established in the facility. RWPs are prepared, issued, and closed according to the approved procedure and establish radiological safety requirements for activities involving exposure to radiation or radioactive materials. RWPs provide information to the worker concerning protective clothing, job/task identification, and special instructions such as radiological hold points. Radiological surveys that supplement RWPs provide information regarding radiation and contamination levels.

The individual requesting work in an area requiring an RWP notifies HP and provides a description of the work in sufficient detail for HP to adequately assess the radiological conditions applicable to the task. HP assesses the radiological conditions and includes information on the RWP concerning protective clothing, and special instructions such as radiological hold points and issues the RWP. Radiological surveys that supplement RWPs provide information regarding radiation and contamination levels. HP closes the RWP upon being notified that the work is completed; after the RWP has expired, or; as deemed necessary by HP due to changing radiological or job conditions.

A General RWP is issued to control routine or repetitive activities, such as tours and inspections or minor work activities, in areas with stable radiological conditions. A General RWP is posted at the access point or centralized location, such as process building Area Control Room. General RWPs are approved for periods not longer than 1 year.

Job-specific RWPs are used to control nonroutine operations or work in areas with changing radiological conditions. The job-specific RWP remains in effect for the duration of the job or until closed by HP. Job-specific RWPs are posted at the access point to the applicable radiological work area.

Radiological surveys are reviewed to evaluate the adequacy of RWP requirements. RWPs are updated or closed and reissued if radiological conditions change to the extent that protective requirements need modification.

RWPs are closed upon completion of the job. HP personnel evaluate the area to ensure that the area has been returned to an acceptable condition (trash and equipment removed). An evaluation is performed to determine if the area should be decontaminated to release the area based on area posting prior to the start of the job.

HP management reviews the RWP closure package to ensure appropriate actions have been taken.

Technical work documents, such as procedures, work packages (information prepared for onsite maintenance work), or job or research plans, may be used to control hands-on work with radioactive materials. When used in lieu of RWPs these technical work documents will contain radiological protection requirements normally specified in RWPs. Radiological Control Hold Points are incorporated into technical work documents for steps that require action by HP to prevent exposures to radiation and radioactive material in excess of action levels, high airborne radioactive material concentrations, or the release of radioactive material to the environment.

Technical work documents used to control radiological work activities receive the same review and approval as an RWP. Technical work documents or continuous HP coverage may be used in lieu of RWPs when approved by the RP Manager. Qualified HP technicians are authorized to provide continuous radiological coverage in lieu of an RWP for short duration (less than one shift) non-complex tasks. When continuous HP coverage is used, requirements normally specified on an RWP are communicated to the worker.

5.3.1.7 Posting and Labeling

Caution signs for Radioactive Material Areas (RMAs), Airborne Radioactivity Areas (ARAs), Radiation Areas (RAs), and High Radiation Areas (HRAs) are maintained as required by 10 CFR 20.1901, 20.1902, 20.1903, 20.1904, and 20.1905, with the interpretation that RMAs located within a posted Contamination Control Zone (CCZ), Contamination Area (CA), High Contamination Area (HCA), ARA, RA, HRA or other posted radiological area are not required to be posted as an RMA since a higher level of control is already required. In addition, the following exemptions to the applicable 10 CFR 20 requirements apply (also see Section 1.8):

1. Containers located in Restricted Areas within the USEC leased area are exempt from container labeling requirements of 10 CFR 20.1904, as it is deemed impractical to label each and every container. In such areas, one sign stating that every container may contain radioactive material will be posted. By procedure, when containers are to be removed from contaminated or potentially contaminated areas, a survey is performed to ensure that contamination is not spread around plant site.

2. Feed, product, and depleted uranium cylinders, which are routinely transported inside the site boundary between facility locations and/or storage areas at the facility, are readily identifiable due to their size and unique construction, and are not routinely labeled as radioactive material. UF_6 cylinders are constantly attended by qualified Radiological Workers during movement.

5.3.1.8 HP Technician Training and Qualification

Training and qualification of HP Technicians and their immediate managers address routine operations and focus on recognizing and handling situations involving both routine and changing radiological conditions.

HP Technician qualification consists of the standardized core course training material[b], facility-specific information, on-the-job training, and passing a final comprehensive written examination. The written examination may be waived for personnel with National Registry of Radiation Protection Technologist (NRRPT) certification. The training program ensures personnel are proficient in radiation measurements, characterization of radiological conditions, release monitoring, and personnel monitoring. Formal remediation protocols have been established.

Entry-level prerequisites are established to ensure that HP Technicians meet minimum standards for physical condition and education. Task qualification for entry level positions may be used until formal training is completed.

Following initial qualification, HP Technicians are requalified every 2 years. The requalification process requires successful completion of a comprehensive written examination or by obtaining NRRPT certification. Personnel who maintain qualifications as HP Technicians satisfy the requirements of Radiological Worker training.

HP Technician Managers shall maintain qualifications as HP Technicians and participate in continuing radiological training programs.

The HP training programs shall be delivered consistent with the training procedures. Training is used to develop the skills necessary to perform assigned work in a competent manner. The training consists of initial, on-the-job, and continuing training. Instructors will meet the minimum qualifications required by Section 6.6.3.6.

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[b] HP Technician course curriculum training modules are listed in Table 5.3-11.

5.3.2 Personnel Exposure Control and Measurement

5.3.2.1 Administrative Control Levels

An Administrative Control Level (ACL) at PORTS of 0.5 rem per year Total Effective Dose Equivalent (TEDE) per person has been established for radiological workers. If an individual exceeds 50 percent of the ACL during a calendar quarter or the ACL in the calendar year, an evaluation is performed by the RP Manager for approval by the General Manager. The evaluation is performed to determine the types of activities that may have contributed to the worker's exposure. This may include, but is not limited to, procedural reviews, work practices, work locations, and job assignments.

Depending upon the conclusions of the evaluation, the individual may be allowed to continue radiological work, however work restrictions may be imposed on individuals whose exposure exceeds published ACLs. Approval for continued work is documented in the evaluation as described in the preceding paragraph which requires approval by the General Manager. Investigations to determine cause, assess the exposure, and document the results are specified by procedure.

5.3.2.2 Radiation Exposure

Both NRC and DOE regulated sources of radiation and radioactive material are interspersed at PORTS. There is also a frequent moving of personnel from a USEC contractor or sub-contractor staff to a DOE contractor or sub-contractor staff. Individual sub-contractors may move back and forth repeatedly during the year. This situation makes separation of personnel exposure between NRC and DOE regulated sources difficult. However, the low cumulative exposures for the site make it feasible not to separate the exposures for personnel being monitored under the USEC certificate.

In order to comply with the personnel monitoring requirements of 10 CFR 20.1502 and the reporting requirements of 10 CFR 19.13, 20.2106 and 20.2206 within the purposes and scopes stated in 10 CFR 19.1, 19.2, 20.1001 and 20.1002, PORTS provides qualified individuals with NVLAP-accredited dosimeters (TLD) and tracks personnel exposure regardless of whether the exposure is from and NRC or DOE regulated source. This applies to internal as well as external exposure. Whenever worker notification is required by 10 CFR 19.13, the individual's "total exposure" while on the Portsmouth site is reported without differentiating between exposure from NRC regulated sources and DOE regulated sources.

To comply with the reporting requirements of 10 CFR 20.2206, PORTS submits an annual report of personnel monitoring information to the Radiation Exposure Information Reporting System (REIRS) Project Manager based on the personnel exposure data base. Dose reports are completed as required for personnel monitored in accordance with 10 CFR 20.1502 (a).

The occupational exposure received by USEC employees, subcontractors, and visitors shall not exceed the 10 CFR 20 Subpart C limits. USEC requires current year exposure history of any occupational worker as required by 10 CFR 20.2104. Personnel declaring pregnancy are advised to keep radiation exposure to an embryo or fetus in accordance with the ALARA principle during the entire gestation period.

Personnel entering an HRA are monitored for beta/gamma radiation. HP personnel shall specify other dosimeters such as finger rings and direct-reading dosimeters where standard TLDs cannot provide the desired information or are not practical. Self-reading or alarming dosimeters are used upon entry into HRAs.

The established personnel monitoring program consists of the following:

1. Establishment of administrative exposure limits that are lower (10 percent of the limits) than 10 CFR 20 Subpart C limits, and in consideration of the chemical toxicity of soluble uranium, limit intakes to 10 mg per week;

2. Provision for dosimeters to measure the external exposure of personnel; and,

3. Analysis of personnel occupational exposure and maintenance of exposure records.

A network of Fixed Nuclear Accident Dosimeters (FNADs) is situated around the leased area. In the event of a criticality, the TLD in the FNADs is processed by a National Voluntary Laboratory Accredited Program (NVLAP) accredited TLD reader. Personal TLDs can also be evaluated for neutron dose.

In addition, security badges contain an indium foil which can be evaluated for neutron activation. If the indium foil indicates exposure to a neutron flux, the TLD can be read and/or biological materials of personnel may be evaluated.

USEC complies with the requirements of Regulatory Guide 8.13 (Rev. 2), "Instructions Concerning Prenatal Radiation Exposure."

5.3.2.3 Internal Dosimetry

The chemical characteristics and retention times of soluble uranium processed at the GDPs are such that renal toxicity limitations are the limiting conditions for health effects. Historical experience at PORTS indicates that exposure derived from low level chronic exposure is less than 2 percent of the annual radiation exposure limits specified in 10 CFR 20.1201.

A bioassay program is employed to confirm the results of radioactive material contamination control and respiratory protection programs. Bioassay results are the primary means of calculating internal doses. Air sampling data is used to trigger special bioassay sampling.

The Committed Effective Dose Equivalents (CEDE) per unit of intake by inhalation from *Federal Guidance Report No. 11*, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," is used to calculate internal dose. Invivo lung counting may also be employed as determined by the RP Manager.

In the event the potential to exceed an intake of 10 mg/week of soluble uranium exists, appropriate measures are employed to assess exposure. A combination of analytical methods such as mass spectrometer, radiochemical separations, alpha/beta proportional counting, and/or alpha spectroscopy are available to assess the extent of the intake and determination of exposure. Isotopic analysis (Alpha Spectroscopy) is only routinely used when performing detailed dose investigations. Table 5.3-8 gives a program description and the analytical methods employed at PORTS.

The routine bioassay program considers the exposure potential to personnel by work location and work activity. The routine sample submission frequencies and administrative control levels are listed in Table 5.3-4. Because chemical toxicity can be limiting when dealing with Class "D" uranium, the uranium action levels have been selected to limit an individual's chronic intake to 10 mg of soluble uranium per week. Investigations are performed when intakes are confirmed to exceed 1 mg of soluble uranium per week.

Personnel whose routine duties require entry into CCZs, CAs, HCAs, or ARAs participate in the routine bioassay program. Personnel who routinely work in these areas have the potential to receive intakes resulting in a CEDE greater than or equal to 0.1 rem CEDE in a year or intakes of 1 mg of soluble uranium per week. Personnel submit bioassay samples, such as urine or fecal samples, and participate in Invivo monitoring as required by the bioassay program.

The biokinetic model presented in International Commission of Radiation Protection (ICRP) 30 models the clearance of UF_6 from the lung. More recent data from the incident that occurred at Sequoyah Fuels, provided information describing the clearance of UF_6 in the first 24 hours. The model used at PORTS to interpret bioassay data and assess initial radionuclide intake is based on the information in NUREG/CR-5566 "Evaluation of Health Effects in Sequoyah Fuels Corporation Workers from Accidental Exposure to Uranium Hexafluoride." This model was used to establish the PORTS action levels listed in Table 5.3-4, because it relates to PORTS bioassay data and accounts specifically for the chemical and physical characteristics of UF_6, the primary chemical form of uranium at GDPs[c]. This model is outlined in PNL-8723, "Internal Dosimetry Technical Basis Manual for Portsmouth and Paducah Gaseous Diffusion Plants."

The technical basis manual was developed in 1993. It provides the scientific, technical, and philosophical bases of the bioassay monitoring and internal dose assessment aspects of the internal dosimetry programs at PORTS. The internal dosimetry program is based on the concept of effective dose equivalent presented in Publications 26 and 30 of the ICRP. Programmatic modifications have been adopted, where appropriate, to demonstrate compliance with the committed dose equivalent limits specified in 10 CFR 20.

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[c] Action levels are calculated using the method presented in Draft ANSI N13.22 (1995), "Bioassay Programs for Uranium." This method assumes the intake occurs at the midpoint of the sampling period for designing the bioassay program.

Table 5.3-10 gives the predicted bioassay results following a 10-mg acute intake of soluble uranium for different sampling times after the intake. These can be compared to the recall and restriction levels for the PORTS bioassay program stated in Table 5.3-4. Calculations assuming a chronic intake of 10 mg/week would yield a projected bioassay result of 391 μg/liter for personnel on a 4-week sampling frequency using the UF_6 biokinetic model. Table 5.3-10 also shows a comparison of the ICRP 30 standard Class D uranium model along with a comparison of the UF_6 model used at PORTS for internal dosimetry calculations.

PORTS calculates an effective Annual Limit Intake (ALI) based on the isotopic contamination ratios found in an individual's primary work facility. Personnel are assigned internal dose based on the calculated ALI, since transuranics at the levels found to date at PORTS (this is discussed in Section 5.3.2.7) are not readily detectable in "spot" urine samples.

PORTS collects "spot" urine samples from personnel rather than 24 hour urine samples. These "spot" samples are adequate to determine compliance with 10 CFR 20.1201(e). Isotopic analysis of fecal samples and 24-hour urine sampling are not routinely performed at PORTS, however, these analyses will be considered when dose assessments exceed 0.5 rem CEDE. The sensitivities of lung counting systems are poor when compared with urinalysis for Class D uranium; lung counting is considered when intake estimates exceed 0.5 rem CEDE.

Personnel participate in follow-up bioassay monitoring when their bioassay results exceed the administrative control levels or as determined by HP. Special bioassay studies are performed as necessary to evaluate the extent of actual personnel exposure in the event of a confirmed release, breakdown of radiological controls, evidence of failure of respiratory protective equipment, or other incidents in which an intake exceeding 0.8 DAC-hr or 1 mg of uranium is suspected. Personnel are selected to submit special urine samples for Tc-99 analysis based on job analysis or air sample data.

PORTS has one area where the potential for exposure to Class Y uranium exists. This is during maintenance of the Calciner located in the X-705 decontamination facility. During maintenance activities involving the breach of the Calciner system, personal protective equipment (including respiratory protection) is required to be worn and special bioassay samples are collected. Breathing zone air sampling is performed during this activity as an alternate means of dose assessment from Class Y materials in the presence of Class D materials. The standard ICRP uranium biokinetic model shall be used to assess dose from intakes of Class Y uranium.

Urinalysis results are reviewed by HP to determine unusual trends. If bioassay sample results indicate an internal exposure that exceeds action levels, additional analyses and removal of the individual from further exposure are considered.

5.3.2.4 External Dosimetry

Persons requiring radiation exposure monitoring per 10 CFR 20.1502(a) wear beta-gamma-sensitive dosimeters (TLDs) which are processed and evaluated by a processor holding current accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Institute of Standards and Technology (NIST). TLDs are exchanged at least quarterly ($\pm$ 2 weeks) unless authorized in writing by the

RP Manager. The TLDs may be supplemented, as appropriate, by other types of dosimeters (e.g., finger rings, direct-reading dosimeters, and neutron dosimeters) and by radiation measurements made with radiation survey instruments. Self-reading or alarming dosimeters are used for entry into HR or VHRAs.

External dosimetry results are reviewed by HP to determine any unusual trends or exposures. If the external exposure status of an individual becomes uncertain, the individual is removed from further exposure until HP determines the exposure status and advises management of any special controls or restrictions to be applied.

5.3.2.5 Radiological Surveys

The radiological survey program consists of routine surveys, work support surveys, and material release activities. Surveys are conducted to support facility activities in a manner that ensures radiological hazards associated with each activity are properly identified, and relative radiation levels and concentrations of radioactive material are determined. Routine survey frequencies are established based on the stability of operation as demonstrated by the consistency of survey results. Radiological surveys for the purposes of establishing personnel protection equipment or for posting requirements are performed by qualified HP personnel.

The routine survey program involves surveys of the facility to determine workplace radiological conditions, effectiveness of contamination control measures, and proper identification and posting of radiological hazards. Areas within the facility are categorized and scheduled for survey commensurate with their relative radiological hazard and contamination potential. Survey frequencies are based on area occupancy, potential for spread of contamination, and process knowledge. The routine survey program is reviewed annually by the RP Manager, documented, maintained, and modified to reflect changes in radiological conditions. Table 5.3-1 provides the basis of the facility contamination survey program for normally accessed areas.

Due to the large physical size of each process building (as much as 60 acres under each roof) and the facility design (control rooms and locker rooms are located inside the Restricted Area), the primary focus of the routine survey program is to quickly identify any breakdown in contamination control. Areas having the highest survey frequency are those that serve as the access, egress, or boundaries between areas of different radiological conditions where the spread of radioactive material is most likely to occur.

Large area wipes may be used to supplement standard smear techniques. If an evaluation indicates that an area wiped is contaminated, a thorough contamination survey is performed. Large area wipes are reported in units of activity/wipe (e.g., <200 dpm/wipe alpha). If results of a large area wipe are less than detectable for the instrument used, the results of the large area wipes are documented as $<$ the minimum detectable activity of the instrument.

Actions for cleanup of the various areas are listed below.

1. Contamination Areas — In the event removable contamination is identified on accessible surfaces exceeding 100 times the levels specified in Table 5.3-2, the area shall be re-posted as an HCA. If access is required to the area, and additional personnel protective equipment will be required for entry, decontamination of the area shall be initiated as soon as practical with consideration of ALARA principles. Decontamination will commence after the evaluation of the associated hazards is complete. The recommendation to decontaminate will come from the evaluation which considers the remaining work evolutions and personnel access requirements.

2. Contamination Control Zones — In the event removable contamination is identified on accessible surfaces exceeding the levels specified in Table 5.3-2, when averaged over an area one square meter or more, the area shall be re-posted as a CA or HCA and actions taken to locate the source of contamination and an evaluation of the radiological hazards posed by the contamination conducted. If access is required to the area, decontamination of the area shall be initiated as soon as practical with consideration of ALARA principles.

3. Noncontaminated Areas — In the event removable contamination is identified exceeding the levels specified in Table 5.3-2, the area shall be re-posted as a CA or HCA and actions taken to locate the source of contamination and an evaluation of the radiological hazards posed by the contamination conducted. If access is required to the area, decontamination of the area will be initiated as soon as practical with consideration of ALARA principles.

Work support surveys are a fundamental element of the RWP process. In-process surveys are conducted as necessary to verify radiological conditions at various points in the work activity and to ensure exposure potentials are maintained in accordance with the ALARA principle. When required by work activities, surveys are conducted to support decontamination efforts and the release of tools, equipment, and waste material from the work area.

Shipments of radioactive materials arriving at PORTS are surveyed by qualified personnel in accordance with 10 CFR 20.1906. Outgoing shipments of radioactive materials are packaged and surveyed by qualified personnel in accordance with 10 CFR 71 and 49 CFR 173.

In the event that radiological surveys indicate radiation levels above the limits for the area, the area is reposted and the cause of the increased radiation investigated.

5.3.2.6 Work Area Air Sampling

To the extent practical radioactive materials are contained and/or confined during processing, transfer, and storage as necessary to maintain intake of such materials by personnel in accordance with the ALARA principle. As appropriate, operations involving readily dispersible forms of radioactive materials are accomplished within enclosures (e.g., process equipment, glove boxes, glove-port hoods, laboratory-type hoods, etc). Portable ventilation units are used in work areas where large portions of process system surfaces are exposed for maintenance.

The air-monitoring program has been established in facilities or areas where production, maintenance, and support activities involve process equipment, or involve hook-ups and disconnects of UF_6 handling equipment. These facilities include areas where airborne radioactivity concentrations may exceed 10 percent

of the derived air concentration (DAC) listed in Table 5.3-5 averaged over 8 hours. The program includes fixed and portable air-sampling equipment, such as permanently installed sampling heads, continuous air monitors (CAMs), portable low and high volume air samplers, and battery powered lapel samplers. Since air sample data is not used as the primary method to determine internal dose, a determination of representativeness of air sampling data is not performed.

Continuous air samplers with a nominal flow rate of 20 lpm (liter per minute) as determined by operating procedures are installed in process and process support facilities where process equipment is operated or maintained. A combination of lapel, low-volume, and high-volume air samplers are used for job coverage air sampling.

Alarming CAMs or HiVols are used to provide on the job coverage of work evolutions where the generation of airborne radioactivity concentration is expected to exceed 8 DAC-hours in a day. Alarming CAMs are used where work evolutions could result in intakes exceeding 24 DAC-hours in a day. The CAM alarm set points are also based on chemical toxicity limits of soluble uranium.

The frequency of exchanging and analyzing sample media for work area samplers is based on historical experience and professional evaluation. Fixed, CAM, and portable air sampler filters are exchanged at least weekly. Due to radon and radon daughters, air samples are routinely allowed to decay for a minimum of 3 days to allow natural occurring radioactive material to decay. Job coverage samples are normally exchanged every shift. Air sample counting equipment and capabilities are stated in Table 5.3-7. Periodically air samples are sent to the X-710 Laboratory for isotopic analysis.

Flow rates through low-volume and high-volume air samplers, as measured by in-line flow rate instrumentation, are checked at the beginning and end of each sampling period. Air sample flow measurement devices are calibrated under standard laboratory conditions at least annually. The NIST traceable standards used have accuracy and precision of 20 percent or better. Lapel samplers are calibrated as described by use procedure.

Grab samples for expedited analysis are collected using annular kinetic impactors (AKI's). AKI samples are taken when evaluating specific job evolutions with the potential for creating airborne radioactivity and determining whether an airborne radioactivity area exists. Personnel use portable contamination survey instruments to perform the expedited analysis of grab samples. Grab samples that exceed 10 percent of the DAC listed in Table 5.3-5 are transported to the HP counting laboratory for more detailed analysis.

A review of historical bioassay data indicates that personnel are not routinely exposed to high levels or airborne uranium. Except where work activities involve cascade component removal, maintenance, decontamination, disassembly, or hook-ups and disconnects of UF_6 handling equipment, concentrations of airborne radioactivity are normally less than 1 percent DAC.

The administrative control levels for job-coverage air samples is 10 percent DAC and a 0.8 DAC-hour/sample. Special bioassay sampling is required when air samples exceed 0.8 DAC-hours and 10 percent DAC. Adjustment for respirator use is considered in determining bioassay monitoring. (Table 5.3-5 summarizes the Portsmouth GDPs airborne radioactivity posting levels and their basis.)

In the event of unexpected or abnormally high airborne radioactivity results (airborne radioactivity concentrations exceed 10 percent of the DAC listed in table 5.3-5 averaged over 8 hours), investigations are undertaken to verify the validity of the result, identify the source of the condition, assess the associated impact, prompt bioassay sampling to determine personnel dose, and if practical, prevent reoccurrence.

Filter media samples have appropriate correction factors applied for alpha and beta particles based on duration of sampling period and environmental conditions of the area being sampled.

5.3.2.7 Physical and Chemical Characteristics of Radioactive Materials Encountered in the Workplace

Activities at GDPs often involve the breach of systems and components containing radioactive material. At times, these systems and components also at times have some amount of removable contamination on their exterior surfaces as a result of the failure of a particular component or other system breach. The amounts of radioactive material on the external surfaces present little hazard from an external radiation standpoint because the major material is uranium which is predominantly an alpha-emitting radionuclide and technetium-99 which is a weak beta emitter.

The primary alpha radiological concern at the GDPs is the internal dose component from inhalation of uranium during a UF_6 release. While the dose per unit of intake is minimal a large intake (10 mg) would result in an estimate of between 20 and 280 mrem CEDE assuming Class D uranium and 0.5% TRU. U-234 alpha energy is 4.8 MeV.

The primary beta radiation is from uranium daughters, Th-234/Pa-234m at 2.28 MeV E_{max} and Tc-99 at 0.29 MeV E_{max}. The exposure pathway for beta is during skin contamination events which yield approximately 0.002 mrem/hour per dpm/cm^2.

Bremsstrahlung and direct gamma radiation are the primary exposure pathways for external dose at the GDPs and are essentially confined to situations dealing with UF_6 cylinders (full and newly emptied) and calibration of instruments. Gamma calibration sources are listed in Chapter 1. Bremsstrahlung radiation is primarily generated when high energy beta radiation interacts with high atomic number atoms, such as uranium in UF_6.

Neutron exposure is confined to areas where large quantities of uranium cylinders are stored. The neutron dose rate in these areas is between 0.1 and 0.3 mrem/hour at 30 cm.

The primary hazard in a GDP is acute exposure of operating personnel to a major release of Class D uranium in the form of UF_6 from the process equipment. Low-level/chronic exposure is possible in certain phases of the process such as feed and withdrawal areas and during process maintenance activities. The airborne radioactivity hazard tends to be the greatest concern, although the feasible dose from inhalation of Class D uranium is quite low.

The principle radionuclide of concern at GDPs is uranium in the chemical form of UF_6. The chemical characteristics of UF_6 result in the formation of HF (hydrogen fluoride) and UO_2F_2 when

introduced to atmospheric conditions. This is an easily visible, irritating buoyant cloud. The olfactory warning properties of airborne exposure are significant because of strong respiratory tract irritation due to HF.

The HF odor is detectable by personnel at very low concentrations (between 0.03 and 0.13 ppm, depending on reference) making an unknown 10-mg intake of soluble uranium unlikely at a GDP facility. Based on stoichiometric calculations, a 1.0 ppm of HF would result in a airborne uranium concentration of approximately 2.43 mg/m^3. Assuming the standard man breathing rate of 1.2 m^3/hour, a 1-hour exposure to an atmosphere 8 to 30 times the threshold detection limit for HF would result in an intake of approximately 2.9 mg of uranium.

The hazards from uranium are twofold - chemical toxicity and radiation. Fortunately, the warning properties and low specific activity combine so that large intakes are unlikely except during inadvertent releases. The chemical toxicity limit is limiting in the lower assay end of the cascade due to the decrease in U-234. The specific activity of depleted uranium is about 0.4 μCi/gram. The specific activity of reactor feed at 4 percent enrichment is still only 2 μCi/gram.

U-234 is the uranium nuclide with the highest dose per inhaled activity. The DAC for the uranium isotopes present in the GDPs (U-234, U-235, U-236, and U-238) only varies from $5x10^{-10}$ μCi/ml to $6x10^{-10}$ μCi/ml. U-234 is also the predominant isotope in terms of total alpha radioactivity in all areas (except at the tails withdrawal points or the process equipment immediately up gradient from that point). U-238 produces slightly more than 1/2 the total activity only in process equipment near "tails." U-235, although enriched to as much as 97 percent of the mass in the past, is never a significant contributor to the total alpha activity. U-235, over the enrichment process range of 0.3 percent to 97 percent by mass, varies between about 1.6 percent to 4.2 percent of the total alpha activity. To determine the actual composition of the uranium, isotopic determinations would have to be performed on nearly every sample.

At different times during operation, limited operations for reprocessing spent fuel elements have resulted in the presence of trace quantities of fission products and transuranic elements at PORTS. During the diffusion process, the transuranic elements occasionally present some control problems in uranium-recovery systems but do not represent a controlling hazard. In the high-enrichment section of the diffusion process, the fission product Tc-99 concentrates to significant levels, requiring special controls during removal and maintenance of equipment. The feed-withdrawal operations concentrate non-volatile uranium decay products in cylinder residues or in liquid UF_6 transfer lines. Nuclides besides uranium are present in the airborne and removable contamination samples in the GDPs. Based on characterization samples collected at PORTS, only the nuclides U-234, U-238, Np-237 and Th-230 ever exceed 10 percent of the total dose from inhalation. The U-234 decay product, Th-230, is present as a trace constituent in most areas of the 40-year-old GDP facilities. Airborne samples are also analyzed for gross beta-gamma contamination due to the potential for uranium daughters and Tc-99. The beta-gamma DAC in use at PORTS corresponds to Th-234 Class Y.

Th-230 has been found to be the predominant low DAC nuclide present. Only Th-230, Np-237, and U-234 are used to calculate the Table 5.3-5 values for airborne alpha radioactivity. In most areas Th-230 is found at less than 0.5 percent of the total activity; however, areas as high as 2 percent have been found

at PORTS. This complicates the internal dose assessment process since Th-230 produces approximately 120 times as much CEDE per unit intake as Class D U-234.

The nuclides Th-228, Pu-238, Pu-239/240, and Am-241 are detectable, but on average, all of these other nuclides would result in less than 20 percent of the total internal dose from inhalation of the mixture by a worker. Disregarding all the nuclides that each contributes less than 10 percent of the total inhalation dose of the mixture is acceptable under 10 CFR 20.1204(g), since the total activity is used for demonstrating compliance, and the total dose from the all disregarded nuclides is less than 30 percent of the total dose. The mixtures used to demonstrate compliance and develop the airborne radioactivity control levels listed in Table 5.3-5 are based only on Th-230 and U-234 for the following reasons:

1. Th-230 and Np-237 have approximately the same internal dose potential;

2. Th-230 is present in most areas; and

3. Np-237 appears in a ratio of 1:5 to Th-230.

For areas with radionuclide mixtures that contain significant amounts of Th/TRU (>8 percent by alpha activity) the area is controlled by the DAC for Th-230. Areas found to contain Th/TRU contamination in excess of 8 percent of the total alpha activity shall have additional administrative controls implemented to minimize the spread of contamination. As an ongoing element of the PORTS radiological characterization efforts, air and removable contamination samples with alpha activity exceeding 1 nCi are sent to the X-710 Laboratory for isotopic analysis to confirm facility contamination ratios.

5.3.2.8 Respiratory Protection

Engineering and administrative controls, including access restrictions and the use of specific work practices designed to minimize airborne contamination or loss of contamination control are used to minimize worker internal exposure. When engineering and administrative controls have been applied and the potential for airborne radioactivity still exists, respiratory protection is used to limit internal exposures. Use of respiratory protection is considered under any of the following conditions:

1. During entry into posted Airborne Radioactivity Areas;

2. During breach of contaminated systems or components;

3. During work in areas or on equipment with removable contamination levels greater than 100 times the levels in Table 5.3-2;

4. During work on contaminated surfaces with the potential to generate airborne radioactivity.

The Respiratory Protection Program follows the requirements of 29 CFR 1910.134 and 10 CFR 20 for use, issuance, training, and qualifications for respirator users. The Respiratory Protection Program is administered by Industrial Hygiene. RWPs specify respiratory protection required for radiological protection

purposes. Respirator use is considered for activities where an individual may be exposed to soluble uranium that may exceed 0.8 DAC-hours or an intake of 1 mg. soluable uranium during a workshift.

In specific situations the use of respiratory protection may be contraindicated due to physical limitations, such as heat stress, or the potential for significantly increased external exposure with approval of the RP Manager (or designee). In such situations, stay time controls to limit intakes are established and continuous workplace airborne monitoring is provided along with expedited analysis of results. Respirator users are trained that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.

5.3.2.9 Occupational Exposure Analysis

As stated in SAR 5.3.2.2 "Radiation Exposure", an annual report of personnel monitoring information is submitted to the Radiation Exposure Information Reporting Systems (REIRS) Project Manager based on the PORTS personnel exposure data base. Since NRC certification, records show that TEDE doses at PORTS are consistently less than the monitoring requirements of 10 CFR 20.

5.3.2.10 Ventilation

Building ventilation systems are described in appropriate building descriptions. In addition to general ventilation systems, portable ventilation units may be employed for short duration jobs when the unprotected worker could potentially exceed 0.8 DAC-hours of exposure. These local ventilation units are equipped with high efficiency particulate air (HEPA) filters and designed to recirculate and discharge room air at low velocities. Activities where these units may be employed are also approved by Safety Analysis and Criticality Safety.

When used for radiological protection purposes, the portable HEPA filtered ventilation units differential pressure is checked per operating procedure. The operating differential pressure range will be per manufacturer's recommendations or as specified in the technical design basis. HEPA filter units are efficiency tested every eighteen months to ensure HEPA filter integrity. Portable HEPA filter units use is specified on the RWP.

HEPA filter systems are tested every eighteen months using a DOP test (or equivalent). The test frequency and the test are in accordance to American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) N-510-89, "Testing of Nuclear Air Cleaning Systems" as it applies to radiological contaminants likely to be found at PORTS.

The average air velocity through openings in uranium sampling and handling hoods containing readily dispersible uranium is a minimum of 100 lfpm (linear ft/min). This velocity is checked at least annually.

PORTS has several permanently installed "glove boxes" for work evolutions that have the potential to generate airborne radioactivity. The glove boxes are maintained with a negative differential pressure of 0.25 inches of water. This differential pressure is maintained anytime that the glove box is in use.

5.3.3 Contamination Control

5.3.3.1 Areas Restricted for Purposes of Radiological Control

Radiological control is provided by controlling access to areas or facilities where radioactive material may be encountered and by requiring that each person who enters those areas or facilities receives the appropriate level of radiological worker training[d]. Access and departure requirements are specified by procedure. Radiological posting is used to alert personnel to the presence of radiation and radioactive materials, aid in minimizing exposures, and prevent the spread of contamination. Where contamination is present, contamination controls are implemented.

Controlled Areas

The Controlled Area is an area outside the restricted area but inside the reservation boundary established such that access can be limited for any reason. The controlled area allows access for members of the general public and radiological workers. Occupationally exposed workers within the controlled area require, as a minimum, General Employee Radiological Training.

Restricted Areas

Each restricted area is conspicuously identified. Unescorted access to Restricted Areas requires, as a minimum, the successful completion of the appropriate level of radiological worker training and, if required, a personnel monitoring device (TLD). (External dosimetry monitoring requirements are stated in SAR 5.3.2.4.) Depending upon the type and extent (or amount) of radioactive material present, Restricted Areas are identified as RMAs, CCZs, Fixed Contamination Areas, Soil Contamination Areas, CAs, HCAs, ARAs, RAs, or HRAs. Restricted Areas are not required to be posted "Restricted Area".

The Restricted Areas at PORTS are predominately confined to production and process support facilities. The support facilities include decontamination, feed and withdrawal, and maintenance facilities that perform work on process equipment that contains radioactive material. Other areas identified as Restricted Areas include UF_6 cylinder storage, onsite laboratories, and radioactive material storage areas.

The restricted areas at PORTS have been identified and documented through the site radiological characterization program. As conditions warrant, Restricted Areas have been established to protect personnel from radiological hazards.

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[d] Personnel are trained commensurate with the hazard per 10 CFR 19; details concerning Visitor Orientation, General Employee Training, and Radiological Worker Training programs are described in Section 6.6.

Radioactive Material Areas

Areas or rooms that contain an amount of radioactive material exceeding 10 times the quantity specified in Appendix C to 10 CFR 20.1001-20.2401 for the material are conspicuously posted "Caution Radioactive Material." As noted in Section 5.3.1, RMAs located within other posted radiological areas are not required to be posted as "Radioactive Material Area" since a higher level of control is already required.

Contamination Control Zones

CCZs provide a boundary to minimize the spread of contamination. CCZs are areas where removable contamination levels are maintained less than the levels specified in Table 5.3-2, but where discrete instances of contamination are likely to be encountered due to the physical size or historical operation of the facility. CCZs are conspicuously posted "Caution, Contamination Control Zone."

Unescorted access to CCZs requires, as a minimum, the successful completion of the appropriate level of Radiological Worker training and a personnel monitoring device (TLD). Equipment and material are monitored prior to exit from CCZs. Personnel exiting CCZs are required to monitor themselves for contamination at the boundary control station prior to exiting, except as noted in Section 5.3.3.5.

The process building cell floors at PORTS are appropriately posted. Except for discrete locations, contamination levels are less than those stated in Table 5.3-2. Due to the unique nature of the cell floor, access to cell floor will be controlled by a RWP or area posting. When work is planned that has the potential to cause contamination levels to exceed those in Table 5.3-2 the area is posted as a CA, HCA or ARA as appropriate. This work is controlled by a RWP as described in Section 5.3.1.6.

In the event that large areas of removable contamination are identified on accessible surfaces exceeding the levels specified in Table 5.3-2, the area will be re-posted as a CA or HCA and actions taken to locate the source of contamination. If access is required to the area, decontamination of the area is initiated as soon as practical with consideration of ALARA principles.

Fixed Contamination Area

Areas with removable contamination levels below Table 5.3-2 values but with surfaces exceeding the values of Table 5.3-2 for total contamination will be controlled as Fixed Contamination Areas (FCA). If the radiation levels exceed 0.05 mrem/hr at 1 meter the area will be posted as a "Fixed Contamination Area."

Soil Contamination Areas

If surveys of soil surfaces conducted in USEC controlled spaces, indicate surface contamination greater than the total contmination levels shown in Table 5.3-2 the area is posted as required by aproved procedures. Prior to and during excavation, surveys are taken of the sub-surface soil to determine extent of contamination. These soil contamination areas are typically a legacy of past DOE operations and considered DOE waste. USEC will not remediate legacy soil contamination areas unless excavation is required in conjunction with a USEC project.

Contamination Areas

CAs are areas where removable contamination level averaged over an area of approximately 1 square meter has been identified as being greater than the levels specified in Table 5.3-2, but not greater than 100 times the levels in Table 5.3-2. CAs are conspicuously posted "Caution, Contamination Area" and personnel access is subject to RWP requirements. Unescorted access to CAs requires, as a minimum,

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the successful completion of Radiological Worker training and an appropriate personnel monitoring device (TLD).

Personnel exiting CAs are required to monitor themselves for contamination after removing their protective clothing and prior to leaving the step-off pad area; except as noted in Section 5.3.3.5. Equipment and materials shall be monitored prior to removal from CAs. If contaminated, the equipment or material will either be decontaminated or contained and controlled as radioactive material.

High Contamination Areas

HCAs are areas where removable contamination levels have been identified as greater than 100 times the levels specified in Table 5.3-2. In addition to the requirements specified for "Contamination Areas," HCAs are conspicuously posted "Caution, High Contamination Area" or "Danger, High Contamination Area" and personnel access is subject to RWP requirements.

Airborne Radioactivity Areas

ARAs are areas where airborne radioactivity may be reasonably expected to exceed 10 percent of the DAC sampled over 8 hours, or a peak concentration of 1 DAC sampled over no more than 1 hour, or soluble uranium concentration exceeds 50 $\mu g/m^3$ averaged over 8 hours. In addition to the requirements specified for "Contamination Areas", ARAs are conspicuously posted "Caution, Airborne Radioactivity Area" and personnel access is subject to RWP requirements.

Radiation Areas

RAs are areas, accessible to personnel, in which radiation levels could result in a person receiving a dose equivalent in excess of 0.005 rem in 1 hour at 30 centimeters. These areas are posted "Caution, Radiation Area" and personnel access subject to RWP or procedural requirements. Unescorted access to RAs requires, as a minimum, the successful completion of Radiological Worker training and an appropriate monitoring device, if required. The RP Manager may exempt the requirement for an RWP in certain RAs as specified in approved procedures.

At USEC facilities, the primary radiation concern is the cylinder yards. When feed cylinders are emptied, short lived radionuclide daughter products of uranium remain in the cylinder. This residual material is called the "heel." After the cylinder is emptied, the cylinder can exhibit a dose rate exceeding 0.1 rem/hour at 30 centimeters for a short period of time. Since the "heel" forms at the bottom of the cylinder and the cylinder is stored on or near the ground, the area where the dose rate exceeds 0.1 rem/hour at 30 centimeters is not accessible. Any cylinder with dose rates exceeding 0.1 rem/hour at 30 centimeters shall not be moved or raised after being placed in the storage area, without continuous coverage by a HP technician.

Freshly emptied cylinders are surveyed and segregated to areas posted as "Radiation Areas" to allow them to decay. Work and entry in these areas is performed under the guidance of an RWP or procedure by qualified Radiological Workers. Cylinders are transported by trained and qualified personnel.

High Radiation Areas

HRAs are areas accessible[e] to personnel where the radiation dose rates are greater than 0.1 rem/hr at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates. HRAs are posted "DANGER, HIGH RADIATION AREA" or "CAUTION, HIGH RADIATION AREA."

In lieu of the requirements of 10 CFR 20.1601(a), each HRA with general area radiation reading greater than 0.1 rem/hour but less than 1 rem/hour is conspicuously posted "Caution, High Radiation Area" and entrance into the area shall be controlled by an RWP[f]. Physical and administrative radiological controls to prevent inadvertent or unauthorized access to High and Very High Radiation Areas is maintained. Minimum requirements for unescorted entry into HRAs include Radiological Worker training and a personal dosimeter. The RWP specifies one or more of the following;

1. Supplemental dosimeters, which may be either a self-reading pocket dosimeter or an alarming dosimeter that integrates the radiation dose rate in the area and alarms when a preset integrated dose is received;

2. Survey meter or dose rate indicating device available at the work area;

3. Periodic radiation surveys at a frequency specified in the RWP performed by HP.

Unescorted entry into HRAs where dose rates exist such that a worker could exceed a whole body effective dose equivalent of 1 rem/hour at 30 centimeters from the source or wall where the radiation is penetrating includes the following additional requirements:

1. A determination of the worker's current exposure, based on primary and supplemental dosimeter readings;

2. Pre-job briefing;

3. Review and determination by HP regarding the required level of HP Technician coverage.

Very High Radiation Areas

Very High Radiation Areas are areas where radiation levels exceed 500 rads/hour at 1 meter from the radiation source or 1 meter from any surface that the radiation penetrates. Very High Radiation Areas are posted "GRAVE DANGER, VERY HIGH RADIATION AREA" and "SPECIAL CONTROLS REQUIRED FOR ENTRY"[g]. Workers are prevented from entry to Very High Radiation Areas when a radiation source is exposed and very high radiation fields are present. In addition, a survey is made prior to the first entry to

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[e] Administrative and physical controls for UF_6 cylinders that exhibit radiation levels greater than 100 mrem/hour at 1 foot are discussed in the "Radiation Area" section.

[f] HP personnel or personnel escorted by HP personnel shall be exempt from the RWP requirement during the performance of their assigned radiological protection duties, provided they are following plant radiological protection procedures for entry into high radiation areas.

the area after the source has been secured or shielded to verify the very high radiation field has been terminated. Facility operations personnel are notified prior to personnel entry to areas where operational or system changes made by operations personnel could result in significantly increased area dose rates. The Plant Shift Superintendent is notified prior to entry into a posted Very High Radiation Area.

Radiography operations are performed by procedure for x-ray radiography as described in the SAR, Chapter 3 or under the license to which the radiography source has been issued. HP verifies radiography boundaries.

5.3.3.2 Contamination Control Boundaries

CCZs, CAs, and HCAs are delineated by a clearly identifiable physical boundary. Step-off pads are provided to minimize the spread of contamination unless otherwise exempted by approved procedures. Personnel monitoring is performed at step-off areas, except as noted in Section 5.3.3.5.

5.3.3.3 Surface Contamination

Loose surface contamination is controlled by posting areas, use of RWPs, protective clothing, surveys and training of personnel. Whenever normally accessible areas of loose surface contamination are found to be above the levels specified in Table 5.3-2, the area is reposted and when appropriate, decontaminated.

5.3.3.4 Use of Protective Clothing and Equipment for Contamination Control

Protective clothing is provided for personnel entering contaminated areas. The type(s) of clothing required is consistent with the individual's work assignment and is dependent upon the type and level of contamination anticipated. With the exception of emergency evacuations, protective clothing is removed prior to exiting the BCS, as specified in Radiological Worker Training, RWP, or area posting. During emergency evacuations, personnel report to designated assembly points and monitoring stations where protective clothing is removed and contamination monitoring is performed.

The protective clothing requirements are specified in the applicable RWP, approved procedures, or area posting; for example, during routine inspection activities, gloves and shoe covers may be the only protective clothing specified. Lab coats are available for use during more detailed inspection activities. Most normal operating and maintenance assignments require shoe covers, coveralls, and gloves. Industrial safety equipment, such as face shields, goggles, and acid suits are available. In addition full-face negative pressure respirators and full-face positive pressure respirators are available. Other NIOSH and MSHA approved devices may also be utilized for respiratory protection.

[g] The only area at PORTS where a Very High Radiation Area can routinely exist is at the Radiation Calibration facility. This facility has established administrative procedures and audible and visible alarms along with entry control devices to prevent inadvertent exposure of personnel.

The alarm set points for laundry monitors in use at PORTS are dependent on background, belt speed, source to detector distance, and detector efficiencies. The limits for laundered anti-contamination clothing shall not exceed 20,000 dpm/100 cm^2 for beta-gamma or 5,000 dpm/100 cm^2 for alpha.

Protective clothing for accident conditions is specified in the Emergency Plan.

5.3.3.5 Personnel Surveys

Personnel survey instruments (except as noted below) are located at step-off pads for use, as needed, by personnel leaving areas controlled for removable surface contamination to areas not controlled for removable contamination. Personnel exiting these areas are required to survey themselves after removing their protective clothing and prior to leaving the step-off area unless background is too high or environmental conditions such as high heat, humidity, etc. interfere. In those cases, the individuals are directed to proceed promptly to an alternate area for monitoring. Equipment used for personnel monitoring along with detection limits is listed in Table 5.3-7. HP and/or management are notified if contamination is detected. These requirements may be preempted during emergency situations as noted in Section 5.3.3.4.

The action levels for personnel monitoring of skin and clothing are the total levels stated in SAR Table 5.3-2. HP and management are notified if contamination is found. Clothing contaminated above the action level is not released unless it is decontaminated to below these levels. If individuals are contaminated above the action level, personnel decontamination is performed as indicated in Section 5.3.3.6. The requirement for monitoring prior to leaving an area may be preempted during emergency evacuations, but monitoring is required at designated assembly points.

5.3.3.6 Personnel Decontamination

HP personnel respond to personnel contamination incidents, assess the consequences, and supervise field decontamination efforts. If contamination is not readily removable, contaminated personnel are decontaminated by qualified personnel. Bioassay sampling may be required based on HP evaluation of the incident. Contaminated individuals may be permitted to leave the facility when it has been determined by the plant medical director (or designee) and RP Manager (or designee) that the remaining contamination does not result in significant exposure and further decontamination is detrimental to the patient. Emergency medical treatment takes precedence over radiological considerations.

5.3.4 Radioactive Material Control

5.3.4.1 Control of Radioactive Material

Containers and equipment containing radioactive material or items contaminated by radioactive material are surveyed, tagged, or labeled in accordance with 10 CFR 20.1904 and 1905, except as specified in Section 5.3.1.7. Procedures govern the requirements and methods for identifying and labeling radioactive material.

5.3.4.2 Radioactive Source Control

Sealed Sources

The Radioactive Source Control Program maintains administrative and physical control of sealed radioactive sources. The Source Control Program establishes custodians throughout the plant and requires semi-annual leak testing, accountability, and control of sealed radioactive sources.

Each sealed source containing more than 100 μCi of beta and/or gamma emitting material or more than 10 μCi of alpha emitting material, other than Hydrogen-3, with a half-life greater than 30 days and in any form other than gas, is tested for leakage and/or contamination at intervals not to exceed 6 months. In the absence of a certificate from a transferor indicating that a test has been made within 6 months prior to the transfer, the sealed source is not put into use until tested.

The test is capable of detecting the presence of 0.005 μCi of contamination on the test sample. The sample is taken from the source or from appropriate accessible surfaces of the container or from the device where the sealed source is mounted or stored in which one might expect contamination to accumulate. Records of leak test results are in units of activity (dpm or μCi) and are maintained for inspection by the Commission.

Leak testing is conducted by HP. If the test reveals the presence of 0.005 μCi or more of removable contamination, USEC will immediately withdraw the sealed source from use and cause it to be decontaminated and repaired by a person appropriately licensed to make such repairs or to be disposed of in accordance with Commission regulations. Within 5 days after determining that any source has leaked, USEC shall file a report with the Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, describing the source, test results, extent of contamination, apparent or suspected cause of source failure, and corrective action taken. A copy of the report is sent to the Administrator, USNRC, Region III, 801 Warrenville Road, Lisle, IL 60532-4351.

The periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources excepted from this test must be tested for leakage prior to any use or transfer to another person unless they have been leak tested within 6 months prior to the date of use or transfer.

Plutonium Alpha Sources

Each plutonium alpha source containing 0.1 μCi or more of plutonium, when not in use, are stored in a closed container adequately designed and constructed to contain plutonium which might otherwise be released during storage.

At least once every 3 months, USEC tests the sources for loss of plutonium in one of the following ways using radiation detection instrumentation capable of detecting 0.005 μCi of alpha contamination: (1) By measurement of the source of potential alpha contamination through surveys of the storage container and areas in which the source is used; or (2) By wiping thoroughly the external surfaces of the source mount, other than the radioactive surface of the source, with a piece of filter paper of high wet strength and low porosity, moistened with a solution which does not attack the mount and, after the paper is allowed to dry, measuring the radioactivity on the paper. Records of test results are in units of activity (dpm or μCi) and are maintained for inspection by the Commission.

If any survey or measurement performed as required above discloses the loss of more than 0.005 μCi of plutonium from the source or if a source has been damaged or broken, the source is deemed to be losing plutonium. USEC shall immediately withdraw it from use and cause it to be decontaminated and repaired by a person appropriately licensed to make such repairs or to be disposed of in accordance with Commission regulations. Within 5 days after determining that any source has leaked, USEC shall file a report with the Division of Fuel Cycle Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, describing the source, test results, extent of contamination, apparent or suspected cause of source failure, and corrective action taken. A copy of the report is sent to the Administrator, USNRC, Region III, 801 Warrenville Road, Lisle, IL 60532-4351.

5.3.4.3 Release of Materials, Equipment, and Facilities

Materials and equipment are not released for unrestricted use unless the contamination levels are less than the levels specified in Table 5.3-2. Contamination surveys are performed on materials, equipment, and facilities to be released from radiological controls.

Use histories are used to supplement surveys of materials or equipment that have inaccessible surfaces. Use histories are summaries of the operational history of the item. Use history information includes the function, location(s) where the item was used, and other relevant evidence to assess the item's potential for internal contamination.

Total radioactivity levels for bulk, aggregate materials, or waste to be released for unrestricted use or disposal are specified in the Radioactive Waste Management Program Document.

5.3.5 Radiological Protection Instruments and Equipment

Radiation dose rate and contamination survey instruments are selected to measure the types and energies of radiation encountered with gaseous diffusion operations. The source term associated with the enrichment process is consistent with little radiological exposure consideration up to a point where the potential for skin exposure is evaluated. As such there is little need for a wide range of instruments. However, survey instruments capable of supporting radiography operations are maintained in inventory.

The primary complement of instrumentation includes alpha/beta count rate and scaler instrumentation plus ion chambers used to evaluate shallow dose and deep dose equivalent readings. Table 5.3-3 describes typical instrumentation available to support the operation of the facility.

The RP Manager is responsible for maintaining adequate quantities of calibrated radiation detection and measurement instruments.

Radiological portable instruments shall be calibrated based on specifications derived from applicable vendors manuals and other nationally recognized guidance as appropriated (e.g., NCRP 112). The standards found in the American National Standards Institute (ANSI) N323 (1978) shall be followed except for Sections 4.6 and 5.1(3). The following requirements apply to all such equipment and instruments:

1. Portable radiation detection and measurement instruments are inspected, maintained, and calibrated at least annually or according to the manufacturer's recommendations, whichever is more frequent, or removed from service.

2. Instruments are calibrated following any maintenance, modification, or repair deemed likely to affect operation before being returned to service.

3. Calibration sources and equipment are within ±10percent of the stated value and have documented traceability links to the National Institute of Standards and Technology.

4. Portable RP instruments that are in use but do not have a built in automatic functional test feature are source checked daily prior to noon that day, or prior to using the instrument if not used on a daily basis. Instruments with the automatic functional test feature that are in use are checked once a week (e.g., hand and foot monitors, half body monitors).

Uranium toxicity hazard takes precedence over the radiological hazard. The survey instruments used are adequate to detect all applicable action levels in Table 5.3-2 and these action levels are adequate to also ensure worker protection against the chemical toxicity of soluble uranium.

5.3.6 Records and Reports

Radiological protection records demonstrate the effectiveness of the overall program and document personnel exposure. Records are maintained in the form required by 10 CFR 20.2110 and are retained as required by 10 CFR 20.2101 through 20.2106. Section 6.8 provides additional information on the Records Management and Document Control program.

5.3.7 Items Addressed By Compliance Plan

This section is implemented as described with exception(s) as listed below. The listing of the exception(s) also contains a brief description of what is currently in place at the plant. The Compliance Plan provides a description of the exceptions (noncompliances), a justification for continued operation, a description of the actions to be taken to achieve compliance and the schedule for completion of those actions.

5.3.7.1 Section Deleted

5.3.7.2 Section Deleted

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Table 5.3-1 Removable Contamination Survey Frequency Basis

Area Surveyed	Survey Frequency
Feed/Withdrawal Stations	Monthly[a]
Uranium Processing Areas	Yearly[b]
Contaminated Maintenance Areas	Quarterly
Contamination Control Zones (CCZ)	Quarterly
Lunchrooms/Breakrooms Within Restricted Areas	Note "d"
Permanent Boundary Control Stations[c]	Weekly
Change Rooms Within Restricted Areas	Monthly
UF_6 Sample Handling Laboratories	Monthly[a]

a. Or following any indication of release.

b. Due to the size of process areas, major access paths and portions of the area are surveyed quarterly. Additionally, localized area surveys are taken following an indication of release and during maintenance activities.

c. When personnel contamination is detected at the BCS, the ensuing follow-up activities include a physical survey of the BCS.

d. Surveys are performed during normal plant working days (i.e., Monday through Friday). Weekends and plant holidays are excluded.

Table 5.3-2 Summary of Contamination Levels

Nuclide[a]	Removable (dpm/100 cm^2)[b]	Total (Fixed + Removable) (dpm/100 cm^2)
U-nat, U-235, U-238, and associated decay products, Transuranics ≤ 2% by alpha activity, Tc-99, and beta-gamma emitters	1000	5000
Transuranic modified materials containing > 2% and < 8% transuranics by alpha activity, Th-nat, Th-232, Ra-223, Ra-224 & U-232	200	1000
Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129, and transuranics ≥ 8% by alpha activity	20	500

a. The values in this table apply to radioactive contamination deposited on, but not incorporated into the interior of, the contaminated item. Where contamination by both alpha and beta-gamma-emitting nuclides exist, the levels established for the alpha- and beta-gamma-emitting nuclides apply independently.

b. The amount of removable radioactive material per 100 cm^2 of surface area is determined by swiping the area with a dry filter or soft absorbent paper while applying moderate pressure and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. For objects with a surface area less than 100 cm^2, the entire surface is swiped; and the activity per unit area is based on the actual surface area. Except for transuranics ≥ 8% by alpha activity, Ra-228, Ac-227, Th-228, Th-230, Pa-231, and alpha emitters, it is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual contamination is within the levels for removable contamination.

The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the level specified in Table 5.3-2. For purposes of averaging, any square meter of surface is considered to be above the level G if; (1) from measurements of a representative number of n of sections it is determined that $1/n \, \Sigma_n \, S_i \geq G$, where S_i is the dpm/100 cm^2 determined from measurements of section i; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds 3G. (G is defined as the levels listed above.)

Table 5.3-3. Radiological protection instrumentation.

Instrument Type	Radiation Detected	Purpose	Typical Range	Minimum Calibration Frequency
Count Rate	β, γ α	Contamination Monitoring	0 - 500,000 cpm	Annually
Dose/Dose Rate	β, γ	Radiation Monitoring	0.2 mR/hr - 5 R/hr 0.1 mR/hr - 1000 R/hr	Annually
Scaler	α, β, γ	Radioactivity Analysis	N/A	Annually
Count Rate	α, β, γ	Personnel Contamination Monitor	N/A	Annually

Table 5.3-4 Internal dosimetry program action levels.[1,2]

Bioassay Technique	Frequency[3]	Action Level	Actions to be Taken
Urinalysis Routine	Monthly	5 μg U/liter	Resample to confirm result and determine intake[4]
	Quarterly	0.5 μg U/liter	
	Monthly or Quarterly	20 μg U/liter	Restrict individual and resample to determine intake[4]
Urinalysis Special	2-6 hours after intake	5 μg U/liter	Resample to confirm result and determine intake[4]
		300 μg U/liter	Restrict individual and resample to determine intake[4]
	16-30 hours after intake	5 μg U/liter	Resample to confirm result and determine intake[4]
		50 μg U/liter	Restrict individual and resample to determine intake[4]
	16-30 hours after intake	Tc-99 6,000 pCi/liter	Resample to confirm result and determine intake
		Tc-99 60,000 pCi/liter	Restrict individual and resample to determine intake
Lung Counting	As Required	>100 μgrams U-235 or 7 nCi Total U	Recount to confirm result and perform urinalysis

1. Internal dose is calculated for all personnel whose average bioassay sample result exceeds 5 pCi/liter in the calendar year.

2. The largest internal dose committed during Calendar Years 1992 through 1996 was 0.079 rem CEDE.

3. Routine sample frequencies determined by evaluation of work location and duties performed. In addition personnel may be assigned a special frequency of every 2 weeks if deemed necessary by HP.

4. When intake is confirmed to be > 1 mg uranium, an investigation is performed to identify the source of the exposure, assess the impact, and if practical, a means to prevent reoccurrence.

Table 5.3-5. DAC and airborne radioactivity posting levels.[3,4,5,6]

NUCLIDE (See Note 1)	DAC	POSTING LEVEL (See note 2)
Gross Alpha based on Class D U-234 and 2% Class W Th-230	1×10^{-10}	1×10^{-11}
Gross Alpha based on Class D U-234 and 8% Class W Th-230	3×10^{-11}	3×10^{-12}
Gross Alpha Class W Th-230	3×10^{-12}	3×10^{-13}
Gross Beta-Gamma (Th-234 Class Y)	6×10^{-08}	6×10^{-09}

Notes:

1. All values are listed with units of μCi/ml.
2. Posting Levels are 10 percent of DAC.
3. DAC - Derived Air Concentration's listed are for the more limiting of 5 rem Committed Effective Dose Equivalent or 50 rem tissue or organ limit.
4. The values above are assumed as worst case, i.e., Th-230 is present in each mixture at the highest concentration per category as described.
5. Area may be posted based on calculated DAC's from actual airborne radioactivity concentration data.
6. The OSHA PEL (chemical toxicity level) for soluble uranium is 50 $\mu g/m^3$ which equates to 2×10^{1} μCi/ml for depleted uranium (tails).

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Table 5.3-7. Radiological protection instrumentation sensitivities.

Instrument	Manufacturer	Use	Detection Limit
LB5100	Tennelec	Air sample counting	α - 4 pCi β,γ - 8 pCi
		Removable contamination counting	α - 20 dpm/100 cm^2 β,γ - 40 dpm/100 cm^2
Alpha Sentry	Canberra	Alarming CAMs	24 DAC-hours[a] 48 DAC-hours
LB1043AS	Berthold	Personnel contamination monitoring	5,000 dpm/100 cm^2 total contamination[b]
PCM2	Eberline	Personnel contamination monitoring	5,000 dpm/100 cm^2 total contamination
Ludlum 12 with GM probe	Ludlum	General purpose contamination monitoring	100 cpm above background[c]
Ludlum 12 with alpha scintillator	Ludlum	General purpose contamination monitoring	100 cpm above background[c]

a. Chronic exposure alarms - is set to alarm at a 800 μg intake assuming no respiratory protection, and assuming depleted uranium.

Acute exposure alarm is set to alarm at a 1600 μg intake assuming no respiratory protection and assuming depleted uranium.

b. The Berthold Monitors are set to alarm with 95 percent confidence upon detection of less than or equal to 5,000 dpm total contamination per detector. The actual detection limits are approximately 3-sigma above background and depend on detector size, efficiency, background, and count time.

c. Personnel are trained in Radiological Worker Training to notify HP when contamination is detected greater than 100 cpm above background. The maximum acceptable background count rate is 300 cpm.

Table 5.3-8. Bioassay program for Portsmouth.

Urine Bioassay Capabilities	PORTS
Number of workers currently participating in urine bioassay program (approximate)	1,500
Workers participation	Selected based on work locations
Normal Urine Sample Collection Time	End-of-shift (Friday afternoon)
Frequency of Urine Monitoring	Monthly & Quarterly [a]
Routine Urine Sample Volume	100 ml
Primary Uranium Analysis Method	ICP Mass Spectroscopy[d]
ICP Mass Spectroscopy Lowest Concentration Reported	<0.006 μg/liter U-235 <0.015 μg/liter U-238
Tc-99 Analysis	Beta Scintillation counting[d]
Technetium Detection Level	10 dpm/ml

Additional Analytical Capabilities

Alpha Spectroscopy[d]	0.1 pCi/sample[bc]
Uranium Alpha with Proportional Counter[d]	40 dpm/liter Total U in urine
Fluorimetry	5 μg/liter Total U
Invivo Lung Counting	0.2 nCi U-235 4 nCi U-238
Transuranic Analyses Performed	When estimated internal dose exceeds 0.5 rem.
Transuranic Analysis Method Alpha Spectroscopy Invivo Lung Counting	 0.1 pCi/sample 0.3 nCi Np-237 or Am-241
Dose assessment software	INDOS (Routine Analysis) CINDY (Developmental and Special)

a. Samples scheduled to be submitted every 4 weeks or every 12 weeks.

b. For the following radionuclides U-234, U-235, U-238, U-236, Am-241, Np-237, Pu-239, Pu-238.

c. Equipment also used for loose contamination and airborne radioactivity samples for characterization efforts.

d. Specific information concerning X-710 Laboratory instrumentation and procedures is stated in Section 5.7.

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Table 5.3-10 10 mg Acute Intake of Uranium

Time After Intake (days)	UF_6 Intake Retention Fraction *	Predicted Result Following a 10-mg Intake (μg/liter)	Time After Intake (days)	ICRP 30 Intake Retention Fraction	Predicted Result Following a 10-mg Intake (μg/liter)
0.17	3.85e-01	3852.0	0.17	2.02e-01	2022.0
0.25	3.16e-01	3163.0	0.25	1.82e-01	1820.0
0.33	2.55e-01	2554.0	0.33	1.63e-01	1633.0
1.00	5.37e-02	536.7	1.00	7.42e-02	741.5
7.00	2.21e-03	22.1	7.00	6.47e-03	64.7
14.00	1.20e-03	12.0	14.00	3.51e-03	35.1
28.00	5.09e-04	5.1	28.00	1.32e-03	13.2
42.00	3.14e-04	3.1	42.00	6.48e-04	6.5
56.00	2.35e-04	2.3	56.00	3.67e-04	3.7
70.00	1.91e-04	1.9	70.00	2.20e-04	2.2
84.00	1.61e-04	1.6	84.00	1.34e-04	1.3
98.00	1.41e-04	1.4	98.00	8.29e-05	0.8

* UF_6 model based on PNL-8723

Table 5.3-11. Health Physics Technician course curriculum

Core Academics

Basic Mathematics and Algebra
Unit Analysis and Conversion
Physical Sciences
Nuclear Physics
Sources of Radiation
Radioactivity and Radioactive Decay
Interaction of Radiation with Matter
Biological Effects
Radiological Protection Standards
ALARA
External Exposure Control
Internal Exposure Control
Radiation Detector Theory

Site Academics

Radiological Documentation
Communication Systems
Counting Errors and Statistics
Dosimetry
Contamination Control
Airborne Sampling Program/Methods
Respiratory Protection
Radiological Source Control
Access Control and Work Area Setup
Radiological Work Coverage
Shipment and Receipt of Radioactive Material
Radiological Incidents/Emergencies/First Aid
Personnel Decontamination
Radiological Considerations for First Aid
Radiation Survey Instrumentation
Contamination Monitoring Instrumentation
Air Sampling Equipment
Counting Room Equipment
Chemical Toxicity of Soluble Uranium Compounds

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5.4 FIRE PROTECTION

Pursuant to 10 CFR 76.35(a)(6) and 76.87(c)(4), this section provides a description of the fire protection facilities and equipment used to protect health and safety and limit danger to life or property from fires. This section also provides an overview of the plant's fire protection program.

5.4.1 Program

The fire protection facilities and equipment are designed to detect, contain, and suppress fires. The major physical components of the fire protection system (e.g., the water supply system, pumps, sprinkler systems, fire alarms, and other firefighting equipment) and the location and operating characteristics of these components are described in greater detail in Section 3.6.

The fire protection program is under the direction of the Fire Services Manager. An experienced fire professional has been assigned as the Authority Having Jurisdiction (AHJ) with the responsibility for the interpretation and application of applicable fire codes and standards. The AHJ is a qualified fire protection professional having a bachelors degree in engineering or a technical curriculum and at least 6 years applicable experience. These requirements are similar to the eligibility requirements as Member grade in the Society of Fire Protection Engineers.

The GDP fire protection programs have been developed in accordance with DOE requirements. These requirements draw heavily on NFPA codes and standards and result in fire protection programs that have provided an acceptable level of fire safety.

The fire protection program will comply with the following standards for modifications to the plant following certification: NFPA 10-1990, Portable Fire Extinguishers; NFPA 13-1989, Standard for the Installation of Sprinkler Systems; NFPA 15-1990, Water Spray Systems; NFPA 24-1992, Private Fire Service Mains; and NFPA 30-1990, Flammable Liquids. Any areas where full compliance with these standards will not be observed will be documented and justified by the AHJ. Any deviations found during future modifications will be documented and justified by the AHJ or corrective action will be taken.

The Plant Operations Review Committee (PORC) provides an oversight and review role in fire protection. The charter of the PORC Committee includes the responsibility to review the nuclear safety issues related to the administrative plans, programs, procedures, and changes thereto for programs such as fire protection. This includes procedures and issues that relate to the Technical Safety Requirements TSRs for fire protection. (See Section 6.2.)

To assist the PORC with its responsibilities, the PORC Chair has established a Fire Protection subcommittee. The membership, structure and responsibilities of the subcommittee are defined in a charter.

5.4.1.1 Description of Fire Hazards

The major buildings on plant site (Process Buildings X-333, X-330, and X-326) are constructed of heavy unprotected steel frame, concrete floors, noncombustible (mostly transite) exterior walls, and

a built up roof on a metal deck roof. Each building is considered a single fire area; sprinkler coverage is provided except for cell housings and surge drum rooms. The sprinkler and water systems are described in Section 3.6. Combustible loading is low (except for the lube oil) and the fire hazards are limited to normal industrial activities. Any exceptions to this are identified in the building survey report or by the building custodian. These include such things as switch gear and transformers, fork lift trucks powered by gasoline or diesel fuel, maintenance welding activities, rotating electrical equipment with oil lubricated bearings and some hydraulic systems. Lightning protection was not installed on the major buildings. However, the buildings are heavily grounded, and lightning has not been a problem for these structures. The power systems are designed to handle lightning.

The two major fire potentials are the quantities of lube oil used in the process buildings and the combustible metal deck roof assembly. Lube oil is a closed system , where the oil is pumped to roof level tanks and then gravity fed via piping, to the rotating equipment and then gravity drained (in pipe) to ground floor storage tanks and pumps. The oil used has a flash point of greater than 400 °F and both the roof level and ground level tanks are sprinklered and diked. The roof deck construction is a typical metal deck roof assembly with a combustible vapor barrier/adhesive, insulation board, built up tar and felt covering, covered with gravel. Sprinklers are provided throughout the buildings primarily in response to the fire hazard of the combination of lube oil and combustible roof assembly.

Process buildings and UF_6 handling buildings are single fire areas. None of these buildings are divided into fire areas by fire doors. Building separation is used as a method of limiting fire spread.

There is no fixed fire detection or suppression in the cell housings. The only sources of combustibles in a cell housing is a single hydraulic oil line that operates a control valves. Due to the high flash point of this oil and the lack of ignition sources in the cell housing, the probability of a fire in the cell housing is very low. However, if this hydraulic oil line did rupture, then pressure indicators as described in Section 3.1.1.6.3 would alert operators to the problem. Also, in the unlikely event of a fire, the UF_6 detectors described in Section 3.1.1.11.2 could detect the smoke generated from the fire.

If the fire was generated from a source exterior to the cell housing, such as a lube oil fire at a process motor, other process alarms would alert operators to the problem. These include alarms for process lube oil system pressure, failure of the motor bearings resulting in compressor deblades, power surges and fluctuations, cascade flow alarms, and seal exhaust alarms. These alarms are described in Section 3.1.1.

A combustible loading study for the Process Buildings was completed in 1997. This study concluded that in general, the lube oil fire as described in Section 4 of this document, is a more severe fire hazard than the storage of combustibles in the buildings.

Combustible storage in the process buildings is considered as part of the hazard evaluation described in Section 5.4.1.2. The existing sprinklers have been determined to be adequate to protect the buildings from fire and prevent structural collapse.

Outside of the process buildings there are three areas where liquid UF_6 is routinely handled. These are X-343, X-342A, and X-344A. In these facilities cylinders are heated to liquify the UF_6. In each of these buildings cylinders are heated by steam in autoclaves. The autoclaves have some hydraulic controls on them. The hydraulic fluids have a flash point of greater than 400 °F. Combustible loading in these noncombustible buildings is low and the buildings are sprinklered. Other fire hazards are limited to typical industrial equipment.

There are no significant quantities of flammable liquids used in the enrichment process. The incidental use of these liquids, primarily for maintenance and support activities, is handled under the guidance of the NFPA 30.

Hot work operations are normal in the maintenance activities associated with the operation of the process. These operations are covered by a permit system. This includes pre-job inspection, fire watch stand by during hotwork, and post-job fire watch to prevent delayed ignition of any combustibles. Fire watches receive both classroom and hands-on fire extinguisher training.

The cable tunnels connecting the switch houses, process buildings, and central control facilities at PGDP and PORTS are similar. The tunnels at PGDP were evaluated in September, 1966 and again in May, 1970 and were determined to be low risk installations, as described below. No major modifications and changes have occurred in these tunnels since the evaluations were performed. The tunnels are approximately seven feet wide and seven feet high and contain approximately 75 cables mainly for control and communications, though some 440 V AC and 250 V DC circuits do exist. All cables are insulated for 600 volts except for communications cables that are located in a separate low voltage tray. All cables have neoprene or PVC jackets which are considered flame retardant. The vast majority of the cables carry negligible currents and do not produce measurable heat. There are very few cable splices.

Cell control functions are located in the ACR, the LCC and the PCF/CCF and are redundant. Therefore, the effect of a loss of cables due to a fire in the cable tunnel would be at most an operational inconvenience. All circuits are protected by fuses or circuit breakers where concern exists for short circuits. All materials of construction used in the tunnels are noncombustible including twelve (six on each side) solid transite cable trays on about one foot spacing. These trays have been maintained clean and free of debris and/or combustibles. Transient combustible loadings are small and limited in magnitude. The only reasonable source of ignition is electrical in origin and its probability is very small.

5.4.1.2 Hazard Evaluation

Fire hazards for process buildings (listed in section 5.4.1.1) are evaluated annually by fire protection engineering staff and are documented in a building survey. The fire hazards evaluation activity consists of two major parts, the annual building surveys and the Chapter 4 accident analysis. A survey is an inspection and analysis with a focus on fire protection. These surveys provide a formal review and periodic evaluation of the occupancy and the fire protection associated with a facility.

Completing a building survey consists of these elements:

- Identify building construction.
- Define fire areas.
- Evaluate fire cutoffs or barriers.
- Determine exposures to the structure or facility.
- Describe building function including occupancy classification.
- Assess ordinary building hazards, such as ventilation and heating systems.
- Discuss processes including any special equipment or special operation.
- Assess special hazards such as flammable liquid processes, high piled stock, and classified electrical installations.
- Review fire protection and installed detection equipment as well as special features of fire protection in the building.
- Develop a list of issues or recommendations for the facility manager regarding fire protection issues. These are tracked to resolution.
- Review emergency egress paths.

Chapter 4 contains these elements:

- Accident analysis including major fire scenarios.
- The effect of the fire protection system in controlling the fire scenarios.
- Toxic and radiological hazards from a release regardless of the initiator.

The hazard evaluations performed in this manner have served the site's fire protection program well in the past by identifying issues and problems, to facilitate the continuation of a successful and safe fire protection program.

Review of the emergency egress paths for the facility is accomplished, using the intent of NFPA 101, Life Safety Code as guidance. The process buildings do not comply with the travel distances due to the size of the buildings. Exit arrangements are adequate because of the low occupancy levels, large number of exits, and fixed fire suppression systems in the process buildings.

5.4.1.3 Pre-Fire Planning

An Emergency Plan implementing procedure requires Emergency Packets to be developed and updated annually by the facility custodian (the person responsible for the facility). These packets, located in the facilities, contain information about the building, the layout, specific hazards and other information

applicable to the individual facility. The building packets are also located in the X-1020 and X-300 facilities to ensure they are available if needed. These packets are reviewed as part of the facility inspection by Fire Services personnel. Emergency Management has overall responsibility for the implementing procedure.

5.4.2 Fixed Fire Suppression and Fire Detection Systems

The plant fire alarm system monitors fire alarms in all important buildings and structures (A listing of these facilities is maintained by Fire Services). Alarms caused by non-fire conditions, such as spurious water flow alarms from pressure surges, are reviewed by Fire Services personnel and identified for maintenance as needed. These alarms are not considered as reportable events as described in Section 6.9. The system includes alarm notification to the X-1007 Fire Station and, as a backup, the X-300 Plant Control Facility. The alarm room at the X-1007 Fire Station is manned by Fire Services personnel. During some emergency situations when the alarm room operator is needed at the incident, the alarm receiving responsibility is transferred to X-300 where backup alarm receiving equipment is continuously monitored by operations personnel. This backup capability includes waterflow alarms from the sprinkler systems, manual pull stations located throughout the site, and other special detection systems such as smoke, heat, and CO_2 discharge. This provides for prompt dispatch of emergency response personnel to investigate and resolve the alarm condition. There is no process-related flammable gas or flammable vapor detection needed and no alarms of this type are on the system.

Manual pull stations are located throughout the site. Operation of a pull station will initiate an alarm at the central alarm receiving location. Typically this alarm is not announced locally. Most facilities have local evacuation alarm stations located at the same location as the manual pull stations. The process buildings do not have local evacuation stations, but do have evacuation alarms that are initiated from a central location.

Fixed automatic fire suppression systems provide the primary means of detection, control, and suppression of fires at the plant. These systems, primarily sprinkler systems, are installed in most of the process-related facilities, including those building areas containing systems designated as Q as defined in Section 3.8. These fixed fire suppression systems are inspected, tested, and maintained on a regular basis in accordance with approved procedures (see Section 5.4.4).

A reliable fire water system with water storage, pumps, and underground piping is provided. This is a looped gridded system, intended to provide minimum outage potential. Fire pumps with water supplies are split (located on opposite sides of the site) to provide maximum reliability. It would require multiple failures to render the fire water system inoperable. Sectional valves are provided throughout the system to permit isolation in the event of a pipe break, and pumping capacities are split to provide greater reliability and redundancy.

Details of these systems are contained in Section 3.6.

Emergency response is provided by the onsite Fire Department (Fire Services). Fire alarms are not transmitted offsite to the area fire departments. The site fire services currently has mutual aid agreements with other fire departments in the county. These agreements are tested during periodic exercises held by

Emergency Management and have been used by offsite departments at their request (typically a few times per year).

5.4.3 Mobile and Portable Equipment

Mobile fire equipment is provided and maintained on site to support fire fighting activities and to back-up the fixed fire suppression systems. This equipment is manned by Fire Department (fire services) personnel and includes a minimum of one 1,000-gpm pumper, one truck with HAZMAT, radiological, and rescue equipment, and one ambulance. This equipment is typically housed indoors and is equipped with the necessary hose, nozzles, breathing apparatus, meters, detection equipment, rescue equipment, and other related equipment. Hose carts are provided in the process buildings to facilitate manual fire suppression efforts.

Self-contained breathing apparatuses (SCBAs) are provided for use by trained personnel in connection with emergency activities including firefighting. Breathing air used in SCBAs meets a minimum quality of Grade D.

Portable fire extinguishers are available throughout the plant, including the process areas. Primarily, the extinguishers in the process areas are Class ABC dry chemical type and Class BC CO_2 and dry chemical type. Size, selection, and distribution of extinguishers determined using NFPA 10 "Standard for Portable Fire Extinguishers" as the guidance document. Extinguishers that are used in the balance of the plant, consist primarily of dry chemical, CO_2, and pressurized water types.

5.4.4 Testing and Inspection

The inspection and testing of fire protection equipment is performed by or overseen by Fire Department (Fire Services). The testing and inspection of equipment is completed in accordance with departmental procedures which include test frequencies that have been developed by DOE-Oak Ridge over the past 40 years. The DOE-Oak Ridge program was based on NFPA inspection frequencies modified to the special situations at the site (such as controlled access, specially trained workforce, supervised systems, onsite Fire Department (fire services) and engineering staff, and extensive operating experience).

The major elements of the inspection program and their associated frequencies are:

- Every three years:
  Trip test dry pipe sprinkler systems,
  Hydrostatically test fire hoses located/stored in process buildings (unless new hose which will be tested within 5 years of purchase).

- Annually:
  Flow test fire pumps
  Calibrate HPFWS pump start switches
  Flow test wet pipe sprinkler systems
  Inspect and flow test fire hydrants
  Test manual fire alarms (pull stations)
  Test sprinkler waterflow alarms

Test supervisory alarm devices including low air pressure, low temperature and loss of power
Flow test pumper trucks
Test SCBA
Operate sprinkler systems control and sectionalizing valves
Test special fire alarm indicators, such as heat and smoke detection systems
Inspect major buildings to evaluate housekeeping, check fire emergency equipment and exit pathways, as described in Section 5.4.1.2.
Hydrostatically test fire hoses on pumper trucks.
Inspect and refold fire hoses located/stored in process buildings.
Inspect exterior of fire water storage tanks

- Quarterly

SCBA air quality checks

- Monthly

Start test fire pumps
Inspect wet pipe sprinkler systems risers
Inspect portable fire extinguishers.
Verify diesel fuel supply to diesel fire pumps
Verify fire water storage tank and water basin level
Assure control and sectionalizing valves to required sprinklers are properly aligned

5.4.5 Staffing and Training

Fire emergency response is primarily handled by the onsite fire department (Fire Services). This is a fully staffed fire department with line officers and firefighters. Emergency response to the entire site is provided by this group under the incident command of the Plant Shift Superintendent (PSS) or his designee. Response capability of the group includes, among other things, fire, rescue, emergency medical, process problems, spills, and confined space rescue.

Fire Services is organized with the Fire Protection Engineers and the Fire Officers reporting to the head of the group. The Fire Protection Engineers are responsible for design review, specification requirements, building surveys, support to other groups, and some fire investigations (see Section 5.4.6.2). The Fire Officers are responsible for daily management of the fire fighters, including the testing program, training, alarm receiving, and emergency response. Fire Protection Engineers are either graduates of a technical program or have at least 6 years experience in fire protection work. Fire Officers have at least 5 years experience in fire service with management and leadership training.

Fire Services personnel are on duty continuously to provide emergency response services and redundant firefighting capability to back-up the automatic fire detection and suppression systems installed throughout the site. The normal minimum scheduled shift staffing is 6 fire services personnel. Minimum staffing for emergency response, as stated in the TSR, is 4 Fire Services personnel, which enables primary entry with sufficient backup. In addition to their response functions, Fire Services personnel are responsible for the testing and inspection of fire protection systems and related equipment.

Fire Services personnel are on-site continuously and are trained and equipped to handle anticipated types of emergencies. Firefighter training is equivalent to the state certified firefighter training curriculum. Emergency medical response personnel meet requirements for state certification as emergency medical technicians (these are usually also firefighters). Qualified instructors provide a range of classroom and hands-on training to maintain standards of performance for all response personnel. Training needs are reviewed annually and the training program modified to meet identified needs.

Training records are kept of the training activities. Training is based on national standard emergency response methodology with site-specific training on issues unique to the site.

Specific training activities include firefighting, hazardous material response, confined space rescue, emergency medical response, radiological emergencies, and rescue. A live fire training facility is onsite to augment firefighting training. Drills are conducted quarterly, as part of the plant emergency plan.

Additional support for the group is provided by an onsite emergency squad. This group is on call for response to assist emergency responders at emergency scenes. Training is provided for the type of activities they may be called upon to perform. Mutual aid agreements with offsite local fire agencies are also in place if so needed. Employees receive initial and biennial fire safety training as part of general employee training on emergency preparedness (see Section 6.6). This includes emergency reporting, facility evacuation, and fire extinguisher familiarization.

5.4.6 Fire Investigation, Permits, and Procedures

5.4.6.1 Impairment Control

Closure of valves on the water system supplying the fire suppression systems is controlled by a written permit system. The valve closure permit system is controlled by Fire Services and therefore Fire Services is notified of the impairment of fire suppression systems. Only groups authorized by Fire Services have the authority to issue permits and operate fire protection valves.

This permit system provides for the notification of the facility custodian (person in charge of the facility) and the PSS of the reason for the impairment, the expected duration of the impairment, the person doing the valving, system restoration time, person restoring the system, and residual partial system impairment (e.g. branch line removed). Compensatory actions will be initiated when TSR-identified process building sprinkler systems are out of service. These may include suspension of hotwork or other hazardous processes, personnel notifications, fire patrols, or other action necessary as determined by the Fire Services Manager. Systems taken out of service for repair are usually returned to service within an 8-hour period; actual required repairs will affect the actual time needed to complete the repair.

5.4.6.2 Fire Protection Engineering

Fire protection engineering support is available on staff to evaluate fire hazards, review changes to maintenance and process systems, and provide inhouse consultation. They also perform the building surveys as described in Section 5.4.1.2.

Fire protection engineers assist in the development of project design criteria, perform design review, and conduct routine engineering consultation as necessary. Fire protection engineering is part of project design teams and routinely reviews project design packages to ensure applicable fire safety issues are addressed. These issues may include construction, egress, facility protection, separation of fire areas, detection systems, and special hazard protection.

All reported fires are investigated using a graded approach. This includes investigations by fire officers, engineers, or by multidiscipline teams as warranted. Results of investigations are considered for distribution throughout the plant to prevent future reoccurrences. See Section 6.9 for the reporting system.

5.4.6.3 Hot Work Permits

A hot work permit system is in place to ensure that cutting, welding, and other hot work conducted in areas not designed or approved for such processes will be done in a manner that is consistent with industry fire prevention practices. Line managers are trained on fire safety and are authorized to write hot work permits. The Fire Services group is notified by the line manager prior to the initial use of a hot work permit. The Fire Services group will maintain a log of these permits and will conduct a field surveillance of work during routine building inspections and when concerns or unusual circumstances exist.

5.4.7 TSR-identified Process Building Sprinkler Systems

The sprinkler systems in the process buildings are designated as AQ in Section 3.8. Although these systems are designated as AQ systems, they are controlled by the fire protection program herein described; the fire protection program provides the level of control necessary to support the accident analysis. The configuration of fire protection systems has been maintained through the use of a plant procedure. This procedure prescribes the requirements for configuration management of fire protection systems and ensures that modifications and repairs to these systems are under the control of the AHJ. In addition, procedures are in place for the operation, testing, inspection, and impairment control of fire protection systems. Together, these procedures and practices provide assurance that fire protection systems and features will perform their intended functions. These procedures will continue to be used to ensure configuration management of fire protection systems.

The fire protection program is controlled by Fire Services, which includes the fire protection engineering staff. This group has oversight of the fire protection systems. System modifications are reviewed by a fire protection engineer for compliance, or equivalent, with the guidance listed in the standards and the particular situation.

Fire protection engineers review plans for system modifications for compliance with the guidance in applicable standards (final approval authority rests with the AHJ). Modifications and repairs are coordinated between maintenance, contractors, and fire services personnel to ensure proper system work is completed. Repairs to a system are done with recognized and approved parts that are commercially available. Repair parts/replacement parts are specified and controlled by plant programs.

Permits for outages and the actual outage of any system are approved by Fire Services. Maintenance is requested through Fire Services, and post-maintenance testing is done or overseen by members of Fire Services.

The operability of the process building sprinkler systems are addressed in applicable TSRs.

5.4.8 Section Deleted

5.5 TRANSPORTATION

10 CFR 76.60(g) applies the requirements of 10 CFR Part 71 to USEC. The packaging and transportation of radioactive materials are conducted in accordance with 10 CFR Part 71. 10 CFR 71 grants general licenses to transport radioactive material in NRC-approved packages and DOT specification containers pending certain conditions. One condition is that the licensee have a quality assurance program approved by the NRC that satisfies the provisions of 10 CFR 71, Subpart H. USEC's Radioactive Material Packaging and Transportation Quality Assurance Program satisfies the provisions of 10 CFR 71, Subpart H, and is provided separate from this application. Plant procedures governing transportation activities ensure that the conditions of 10 CFR 71 are satisfied.

5.5.1 Section Deleted

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5.6 CHEMICAL SAFETY

United States Enrichment Corporation (USEC) operations at PORTS require radioactive, hazardous, and toxic chemicals to support the basic process of uranium enrichment. The enrichment process uses, consumes, combines, and manufactures various hazardous, flammable, reactive, and toxic chemicals. Pursuant to 10 CFR Part 76.87(c)(11), the TSRs include appropriate references to address chemical safety. The Safety Analysis Report (SAR) describes the technical basis program requirements for chemical safety, the integration of chemical safety with uranium enrichment operations, and describes the management systems used by PORTS for chemical safety. The TSRs identify those requirements for control of chemicals and commits to the chemical safety controls described in this section.

5.6.1 Introduction

The PORTS chemical safety controls are limited to non-radiological materials. Radiological materials, safety analyses, and the toxicity of uranium are addressed in Chapter 4 and Section 5.3.

Chemical safety at PORTS consists of the integration of environmental, safety and health management systems to address chemical safety. Chemical safety controls are designed to mitigate the adverse effects of the toxic materials used in the enrichment process to workers, the public and the environment. To achieve this objective, safety analyses, process hazard analyses, and industrial hygiene and safety programs are utilized. Chemical safety utilizes existing plant programs rather than developing new or specifically tailored programs. These referenced programs may or may not contain direct chemical safety references.

Section 6.1 identifies the roles and responsibilities for the Safety and Health program, including chemical safety, receipt and control of hazardous materials, environmental matters, and fire protection. Chemical safety incorporates technical and administrative controls to manage risk. In Section 5.6.13, the chemical safety control strategy is discussed further and the additional controls and requirements utilized to protect workers and the general public from chemical hazards are identified.

5.6.2 Section Deleted

5.6.3 Operating Procedures

Plant procedures are prepared in accordance with the requirements of a formal procedure system. The procedure process and program, described in Section 6.11, is applicable to all plant procedures.

Alarm response procedures are written for those alarms that alarm remotely in addition to locally. All the alarms are present to aid in the detection and mitigation of releases for production purposes only, and are not credited with that function for the purposes of the accident analysis.

5.6.3.1 Sitewide Safety Program Procedures

Safety and health program procedures, applicable to the entire site, are processed in accordance with the plant procedure program discussed in Section 6.11 and are maintained in controlled Safety and Health Manuals.

Industrial hygiene and safety programs used for chemical safety and implemented by sitewide program procedures include:

- Lockout/Tagout (LO/TO) Program
- Confined Space Entry Program
- Safety and Health Work Permit Program
- Hot Work Permit Program
- Personnel Protective Equipment (PPE) Program
- Signs/Labeling/Tagging Program
- Safety Training Programs

These safety and health programs apply to chemical safety as described in the program implementation documents.

5.6.4 Training

Employee training is coordinated by a dedicated training organization. The plant training programs, as described in this section and Section 6.6, apply to chemical safety. The Training group provides basic plant entry training, general employee training, and technical job specific training as required by plant operations.

Cascade operators, chemical operators, maintenance, management, and emergency response personnel have pre-requisite training requirements needed for initial job qualification.

Personnel who operate, maintain, manage, handle, and have emergency response duties for the chemicals addressed in Section 5.6.13.3 under the Process Safety Management (PSM) program are certified by management as being adequately trained for the particular chemical system or related activity. This training supplements the plant training program and occurs at the job-specific level. Requirements for employee job-specific certification are those required by 29 CFR 1910.119(g).

Non-PORTS contractor (typically construction) personnel receive plant access training and plant- specific safety training by PORTS personnel prior to starting work. The contractor or the contractor-designated Safety and Health Officer has the contractual responsibility for internal contractor employee training. This training, performed by the contractor, is supplemented by PORTS approval of the contractor's Safety and Health Plan. If construction activities interface with chemical systems, appropriate training and guidance will be required. This training and job review is provided by PORTS representatives.

5.6.5 Maintenance and Inspection

Maintenance and inspection programs are summarized below and described in detail in Section 6.4, Section 6.3, and in the QA Program description.

Maintenance and inspection requirements and criteria for chemical systems are developed by Engineering in conjunction with the specific facility maintenance organizations, the manufacturer's recommendations, and the SAR. These chemical safety requirements are based on the TSRs, the SAR accident analysis, and/or the Process Safety Management (PSM) program requirements for a particular chemical system and the manufacturer's recommendations.

5.6.5.1 Calibration and Inspection

Specific calibration and inspection requirements are based on operating characteristics, past operating experience, system operating environments, and manufacturer's recommendations.

Maintenance, including calibration and inspection, of chemical systems that affect the assumptions or conclusions of the accident analysis is implemented in accordance with the TSRs and/or accident analysis conditions for a particular system.

Maintenance of chemical systems is performed in accordance with the facility maintenance programs that include the PSM Mechanical Integrity program requirements for those systems covered under PSM. These facility programs are based upon calibration and inspection requirements from operational experiences and characteristics of the system.

Facility maintenance personnel are responsible for calibrating to the tolerances and inspection criteria identified by the TSRs, the PSM Mechanical Integrity program, and the maintenance work package requirements.

5.6.5.2 Maintenance Work Packages

Maintenance work packages are prepared to provide the necessary technical and safety guidance for maintenance activities as described in Section 6.4. These work packages are applicable to chemical systems and equipment. Supporting maintenance procedures are subject to the requirements of the plant procedure program described in Section 6.11.

5.6.5.3 Preventive Maintenance and Quality Considerations

Manufacturer's recommendations are used as guides for preventive maintenance on specific chemical systems and equipment. If operational experiences or system characteristics indicate a need for a different preventive maintenance schedule, the preventive maintenance baseline can be changed after appropriate review. Equipment installed or maintenance services provided by contractors for chemical systems are tested and inspected by the contractor as required by the contract for that project. Independent inspection and testing is also performed by PORTS personnel.

Independent overview of maintenance activities on chemical system hardware and requirements are addressed by the QA Program and Configuration Management (CM) programs, as applicable. These independent overview programs include:

- Procurement Quality Requirements
- Construction Inspection
- Testing and Pre-operational Inspection
- Pressure Vessel Inspection
- Crane Inspection
- Operational Readiness Review and Pre-Startup Safety Review Programs
- Plant Operations Review Committee (PORC)

The operational readiness review process is conducted in accordance with program implementing procedures utilizing a graded approach. The scope of the readiness review is decided by the readiness review committee depending on the issue and system being reviewed and the safety concerns present.

Deficiencies associated with maintenance activities are dispositioned in accordance with the QA Program requirements, and include Problem Reports, when required.

5.6.6 Configuration Management

The CM program is described in Section 6.3 and is applicable to chemical systems as described in the QAP. These systems are specifically identified and the structures, systems and components (SSCs) are classified using a graded approach as described in the QA program. Engineering, as the design authority for the plant, administers the CM Program. The plant CM Program includes an organizational structure and administrative processes and controls to ensure that accurate, current design documentation is maintained that matches the plant physical configuration.

Revision or modification to a chemical system is initiated via an Engineering Service Order that initiates the design process and includes a 10 CFR 76.68 screening. In accordance with 29 CFR 1910.119(I), a pre-startup safety review is performed for new or modified facilities when the change is significant enough to require a change to process safety information. The pre-startup safety review is an independent review to address the readiness of the system hardware, associated hazard controls, personnel (including required training), and procedures. If the change does not require a change to process safety information, a pre-startup safety review will not be performed.

5.6.7 Emergency Planning

Emergency Planning is performed by the plant Emergency Management group. The PORTS Emergency Plan, contained in this Application, applies to chemical safety concerns. Section 4 of the Emergency Plan outlines the roles and responsibilities of personnel during an emergency and Section 5 describes the emergency response measures in detail, including onsite and offsite protective actions.

PORTS personnel who have emergency response assignments or duties associated with chemicals discussed in Section 5.6.13 are certified as being adequately trained to respond to chemical and operational upsets per 29 CFR 1910.119(g) requirements.

Chemical system operators, in compliance with the PORTS "See and Flee" policy, are not expected to participate in emergency response activities for chemical releases. The policy is for employees to promptly move to a safe location away from the immediate release area. Mitigating actions, as described by procedure, may be performed during evacuation from the immediate release area if they do not hinder safe egress. Personnel outside the immediate release area may perform mitigating actions, as described by procedure, prior to evacuation. If facility procedures direct an employee response to a minor spill, an employee can implement the facility Spill Prevention Control and Countermeasure (SPCC) Plan and/or Best Management Practices (BMP) Plan after "See and Flee" requirements have been accomplished and the area may be reentered. The SPCC and BMP incorporate the handling, storage, spill containment, cleanup, security, and spill reporting requirements for oils and hazardous and/or toxic pollutants as per EPA regulations 40 CFR 112 and 40 CFR 125, Subpart K, respectively.

5.6.8 Event Investigation

The Event Investigations program and process, as described in Section 6.9, applies to chemical safety. Event investigations are conducted in accordance with plant procedures. The level of investigation is based upon severity and significance of the event and regulatory requirements.

Occupational injury and illness investigations related to chemical safety are part of Industrial Safety programs. Investigations are conducted in accordance with Occupational Safety and Health Administration (OSHA) program requirements.

5.6.9 Audits and Inspections

Formal plant audit responsibilities are assigned to the QA organization. In addition, internal organizations have monitoring programs, assessments, and reviews as required by program implementation procedures. The PORTS audit and assessment program is described in Section 6.8 and is applicable to chemical safety.

5.6.10 Quality Assurance

Chemical safety is addressed by the QA Program utilizing a graded approach.

5.6.11 Human Factors

Human factors design responsibility for facility and system design at PORTS is assigned to Engineering with specific technical assistance from Industrial Safety personnel. The process hazard

analysis, prepared for OSHA PSM regulated chemicals, includes ergonomic considerations at the specific work site.

5.6.12 Detection and Monitoring

Technical descriptions of the methods and equipment used for detection and monitoring of releases are discussed in Sections 3 and/or in the process hazard analyses prepared for the OSHA PSM standard for the specific facility or system evaluated for chemical safety. For chemical systems, the functional descriptions are contained in Chapter 3.

5.6.13 Chemical Safety Control Strategy

The chemical safety control strategy at PORTS first requires that the chemicals used be identified and the listing of chemicals used kept current. Then, the chemicals are categorized by potential chemical risk. In order of decreasing risk and decreasing significance, the chemical hazards are addressed by the Chapter 4 accident analyses, by the Process Safety Management program, and by the industrial hygiene and safety programs for chemical hazards.

5.6.13.1 Identification and Inventory Control

Three processes are used to identify hazardous or toxic chemical use at PORTS.

The first process identifies and inventories chemicals used on the PORTS site. This process ensures that chemicals used on site are appropriately addressed for safety. The process includes:

- Purchase requisition reviews.
- A listing of chemicals used by PORTS.
- Material Safety Data Sheet (MSDS) library, upgrades, and distribution services to the site.
- Identification of new chemicals for the review process.

The second process is the formal Engineering Service Orders (ESO) required for modifications to existing systems. The ESO process provides a mechanism that identifies new or revised usages of chemicals, chemical processes, and/or associated possible logistics that require engineering involvement. An ESO may not be required unless physical modifications or updated engineering evaluations are needed. If any changes to hazardous chemical inventories or locations exist as a result of an ESO for a new, modified, or decommissioned facility, process, or storage location, appropriate chemical safety review will be applied to address regulatory requirements. Any physical change to the plant, including inventory limits and changes of location for hazardous chemicals, will be evaluated in accordance with the requirements of 10 CFR 76.68.

The third process is associated with contractors on site. When work is to be performed by contractors, a review of the contractors' safety and health program is conducted to identify the presence of hazardous and toxic materials to be brought onsite by the contractor. MSDS for these chemicals are provided by the contractor, and the chemical is entered into the plant centralized listing.

The three processes described above identify chemicals to be evaluated and controlled at PORTS.

5.6.13.2 Chemicals Addressed By Accident Analysis

The Chapter 4 accident analyses address risks associated with UF_6 and HF. The analyses identify engineered and administrative controls required for safety.

Uranium hexafluoride (UF_6) is the most abundant hazardous material on site. Chapter 4 provides an evaluation of accidents that involve the release of UF_6. The accident analysis considers both radiological and toxicological hazards from the UF_6 releases. The HF which evolves from a UF_6 release is considered as one of the effects of a UF_6 reelase and as such is addressed in Chapter 4.

5.6.13.3 Chemicals Addressed by Process Safety Management

A number of chemicals are used at PORTS that are managed in accordance with the requirements of 29 CFR 1910.119. The PSM Program manages these hazardous chemicals in a manner that prevents impact to workers or the public.

In addition to the chemical process hazard analyses described in Chapter 4, the PSM Process Hazard Analyses (PHA) are used to identify and manage chemical risks and impacts to plant systems or operations, workers, and to the public. Specific job-related risks and the technical and administrative controls used for risk management are addressed in the PHAs.

The chemicals used at PORTS that are in excess of the threshold limits identified in 29 CFR 1910.119 are chlorine, chlorine trifluoride, hydrogen fluoride, and fluorine. The chemicals and locations of use are described in Chapter 3 and summarized in this section. In addition, 29 CFR 1910.119 requires annual certification for operating, supervising, maintenance, handling, and emergency response personnel that are involved with chemicals in excess of PSM threshold limits.

The 29 CFR 1910.119 covered uses, locations, typical working inventory and storage quantities at PORTS are:

- Chlorine

X-611E Water Treatment Plant Chlorination Upgrade	6000 pounds

- Hydrogen Fluoride/Fluorine

X-342 Fluorine Generation Complex (2)	16,100 pounds

- Chlorine Trifluoride

X-330/X-333 Feed Station	3,400 pounds
X-742 Storage Facility	3,750 pounds

PORTS is in compliance with 29 CFR 1910.119 for completion of the Process Hazard Analysis (PHA). As noted above, the current safety requirements for these systems are discussed in Chapters 3 and 4.

Previously undisclosed or newly revealed chemical safety concerns identified by the PHA preparation process are processed through a 10 CFR 76.68 review for disposition of the concern. This 10 CFR 76.68 review will determine if new TSRs need to be developed.

5.6.13.4 Industrial Safety Program Managed Chemicals

Hazardous and toxic chemicals, not covered by the programs discussed above, are effectively managed using Industrial Safety programs. For Industrial Safety program managed chemicals, there is not a "threshold quantity" qualifier. To address these chemicals, the Industrial Safety programs provide the necessary protective barriers and controls enabling safe use of these chemicals.

Commercial chemicals have varying toxicity and hazardous ranges and categories. Because chemicals can be used across the site in various manners, the Industrial Safety program applications to chemical safety are general in nature and based on industry accepted standards and regulatory requirements for controlling occupational exposures. To address the potential exposure risks associated with Industrial Safety program managed chemicals, PORTS uses chemical review programs, program procedures, and Material Safety Data Sheets (MSDSs). Implementation of these Industrial Safety

programs provide employee protection from hazardous chemicals during daily operations and emergency response.

5.6.14 Multi-Occupancy of the PORTS Site

The creation of USEC resulted in a lease agreement with the Department of Energy (DOE). USEC leases from the DOE certain operating segments and certain support facilities of the original PORTS Gaseous Diffusion site. The remaining site sectors are used by DOE environmental restoration contractors and sub-contractors, the Ohio Valley Electric Company (OVEC) - a supplier of electrical power to PORTS via DOE and the Ohio National Guard. DOE provides information regarding any hazardous chemicals used by these "third-parties" that could impact PORTS nuclear operations. No materials have been identified by DOE as potentially impacting PORTS operations.

5.6.15 Section Deleted

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5.7 ANALYTICAL SUPPORT

5.7.1 Analytical Laboratory Mission and Organization

The primary missions of the laboratory are to supply quality analytical services, to provide process support for equipment and systems, and provide technology recommendations. In addition, the laboratory may provide analytical services to the DOE and other offsite customers under contract agreements. Analytical services provided utilize existing facilities and analytical techniques to process trace and low level radioactivity samples bounded by the limits in Table 1-3.

Organizationally, the laboratory is divided into sections according to functional responsibilities. The sections may be subdivided into laboratories to further define specific areas of responsibilites. Laboratories provide a wide range of analytical services including organic, inorganic, spectrochemical, and radiochemical analyses to plant customers including support of nuclear safety and environmental radiological protection activities.

5.7.1.1 Uranium laboratories

These laboratories perform chemical and isotopic analysis for cascade, chemical, utilities, and feed and transfer functions within Operations. Included in these services are product specification analyses, cascade control measurements, Nuclear Criticality Safety (NCS) limitation analyses, uranium accountability analyses, and mass spectrometry isotopic analysis for barrier performance evaluation; operation of the uranium chain-of-custody facility for the laboratory; preparation of uranium hexafluoride standards; vent monitoring and cell negative testing for Operations (all shifts); ASTM specification testing of materials for Operations; and High Efficiency Particulate Air (HEPA) filter testing.

5.7.1.2 Environmental, Safety & Health laboratories

These laboratories routinely perform inorganic, organic, asbestos and radiochemistry analyses and physical properties measurements of environmental compliance, waste management, industrial hygiene, health physics, industrial safety samples. Services also are provided to Operations and Engineering. Environmental chain-of-custody and sample archive and return operations are conducted within the laboratory. Major plant programs supported by these laboratories include bioassay (urine), National Pollutant Discharge Elimination System, personal air monitoring, continuous vent monitoring, environmental monitoring, HP Survey programs, potable water testing, Recirculating Cooling Water testing, and Resource Conservation and Recovery Act waste characterization.

5.7.1.3 Process Technology & Non-Destructive Assay laboratories

These laboratories provide technical support for continuous vent monitoring, compressor vented cavity line cleaning, infrared analysis, failure analysis, scanning electron microscopy, transmission electron microscopy, x-ray diffraction, portable infrared analyzer (PIRA) buggies, conditioning services, trapping materials, video scoping, materials testing, and NDA support.

Laboratory personnel provide statistical support to the laboratory, Nuclear Material Control & Accountability (NMC&A), and the plant at large, as well as administering internal and external laboratory quality control (QC) programs. The Laboratory QA Officer provides QA oversight of the laboratory.

Furthermore, laboratory personnel coordinate the maintenance of the X-710 facility, including maintenance to and monitoring of laboratory hood air flows; procure chemicals and maintain the chemical distribution and reagent preparation facility; and provides in-house laboratory instrument maintenance and equipment fabrication services.

5.7.2 Laboratory Personnel Training and Qualification

Laboratory personnel include a balanced mix of professional scientists and technicians. Professional personnel typically have a bachelor's degree or higher in a scientific or technical field. With few exceptions, entry level technicians have an associate degree or higher in a scientific or technical field. Job experience may be substituted for educational requirements for both professional and technician-level positions.

All laboratory personnel are required to complete the General Employee and Compliance Training modules that are developed by the training organization, as described in Section 6.6.

On-the-job training (OJT) and qualification requirements are documented in standard operating procedures. New or inexperienced analytical laboratory employees are provided appropriate safety and technical training, including instruction in recordkeeping and in the analytical methods they are assigned. An experienced technician or laboratory supervisor works with the trainee until proficiency in the method or procedure is observed as defined in the SAR Section 6.6. Where applicable, the qualification process for quantitative analytical work includes assessing performance with control samples. Qualification folders are maintained for each analyst.

Supplementing lab OJT is required reading. Required reading provides an administratively controlled means of conveying pertinent information related to laboratory personnel. Required readings may include newly revised or developed procedures, lessons learned, guidance from compliance organizations, and other material as determined by management.

Laboratory personnel receive initial and follow-up training in the Chemical Hygiene Plan. In addition, they receive training in basic nuclear criticality safety and in radiation worker safety as defined in the SAR Section 6.6. Monthly safety meetings reinforce safety principles. These meetings serve two purposes: to alert laboratory personnel to potential safety concerns, and to familiarize personnel with proper procedures in the unlikely event an emergency arises.

Laboratory workers performing tasks for which NCSAs apply also receive job-specific NCS training, which incorporates specific requirements of Nuclear Criticality Safety Approvals (NCSAs) that apply directly to their work. In addition, laboratory personnel working in radiological controlled areas take Radiological Worker training to acquire knowledge and skills for safety within radiologically controlled areas.

5.7.3 Control of Analytical Processes

Systematic control of the processes involved in analyzing the samples is required in order to generate reliable results. To achieve the desired level of competence, the Portsmouth Analytical Laboratory uses the following major types of planned and systematic controls:

- Analyst training and qualification (previously discussed);
- Implementation of sample tracking and accountability and/or chain-of-custody procedures to ensure traceability and integrity of samples and data;
- Use of NIST or other certified standards for instrument calibration and day-to-day quality control;
- Implementation of routine and special internal controls;
- Calibration and routine maintenance of measuring and test equipment;
- Minimization of Cross-Contamination;
- Implementation of standard operating procedures and standard analytical methods;
- Participation in external control programs;
- Participation in certification and accreditation programs;
- Statistical evaluation and control charting; and
- Surveillance, problem reporting, and corrective actions.

5.7.3.1 Chain of Custody/Accountability

As appropriate, chain-of-custody protocols are implemented from the point of sampling, and are maintained after samples are relinquished to laboratory personnel through the analytical process and subsequent storage until customer pickup or disposal. Appropriate protocols are likewise implemented to track accountability samples from the point of origin through the analytical process and through disposal. A laboratory information system is used for sample tracking.

ES&H samples are received by the laboratory at an ES&H sample receiving facility. Trained personnel subject the incoming samples to a formal sample acceptance process. Samples that meet the acceptance criteria are officially transferred to the laboratory by signature on a chain-of-custody record. The samples are logged onto the laboratory information system and are then distributed to the appropriate laboratories. Samples that do not meet acceptance criteria are returned to the customer, or, if the customer requests, they are analyzed and the results reported with qualifying statements indicating what caused the samples to fail the acceptance criteria (e.g., improper preservation, indication of leaking, outside regulatory holding time).

Accountability samples are transferred to and from the laboratory with tracking documents, and with the exception of U-tubes, miscellaneous solids, miscellaneous liquids, and test samples for barrier quality, are tracked on the site computer system, Dynamic Nuclear Material Control and Accountability System (DYMCAS). In addition, all accountability samples are tracked from the time of receipt until transfer from the laboratory. The purpose for tracking is to ensure that ^{235}U levels do not exceed Material Balance Area (MBA) licensed possession limits and to facilitate inventory. Like ES&H samples, accountability samples are received by the laboratory at a designated uranium sample receiving facility.

5.7.3.2 Traceability of Standards

The laboratory generates internal QC samples and standards using materials traceable to NIST or other reliable source materials. The samples are then submitted to the laboratories at an established frequency. Standards prepared by the laboratory are used to calibrate instruments and to provide both known and blind QC control samples for analytical sample batches.

5.7.3.3 Routine and Special Internal Quality Control Programs

Good measurement practices for day-to-day QC include use of matrix spikes, matrix spike duplicates, replicate samples, check samples, and various other internal controls. In addition to such known QC samples included with routine analyses, the laboratory also provides blind and double-blind QC samples as further performance checks on analyses. (A blind QC sample is one in which the analyst is aware that the sample is a control, but does not know the analyte concentration. A double-blind QC sample is one in which the analyst is not aware that the sample is a control.)

5.7.3.4 Calibration/Calibration Verification

Equipment is calibrated or a calibration verification performed daily before use according to documented methods or protocols. Thermometers are verified annually to read within a required accuracy range. Analytical balances are calibrated and serviced annually by an independent vendor using NIST-traceable checkweights. Subsequently, the calibration of balances is verified daily before use with NIST-traceable checkweights. Automatic pipettors are calibrated monthly according to standard operating methods.

Equipment calibration or calibration verification is performed according to requirements in regulatory methods or other approved methods. When available, material from two independent sources is used to calibrate and to verify calibration of an instrument. Additional information concerning equipment calibration frequency is provided in the section on analytical methods.

Table 5.7-1 lists analytical instruments and calibration/calibration verification frequencies for the radiological analyses described in this section.

5.7.3.5 Instrument Maintenance/Hood Maintenance

In-house service crews provide instrument maintenance support for analytical equipment. In some cases service contracts for hardware and software maintenance support are maintained with a vendor. Finally, vendors may be called in on an emergency basis to service some equipment. It is a laboratory requirement to maintain a maintenance logbook for each piece of analytical equipment and to note both routine and special maintenance given to each instrument.

Laboratory fume hoods are serviced by an in-house maintenance crews. Face velocity measurements are taken on all fume hoods per the requirements of the Chemical Hygiene Plan. Individual hood velocity readings and flow calculations are recorded and maintained. The test results that do not meet the minimum flow requirements based on the intended use as determined by area supervision are flagged out. Hoods that fail the performance test are tagged out and scheduled for corrective maintenance. Performance data posted at each fume hood list the date of test, average face velocities, and hood class (I, II, III, or radiological) based on the intended use of the fume hood.

5.7.3.6 Minimization of Cross-Contamination

Samples with contamination are identified as required by Section 5.3 of the SAR prior to delivery to the analytical laboratories. Acknowledging the possibility of cross-contamination, analytical laboratories use good laboratory techniques to minimize cross-contamination of samples. These practices include glassware cleaning techniques, analysis of blanks and other quality control samples, storage and preparation of contaminated samples in designated areas, and contamination control activities. Specific methods are documented in analytical procedures.

5.7.3.7 Standard Operating Procedures and Standard Analytical Methods

The PORTS Analytical Laboratory performs analyses according to regulator's methods [Environmental Protection Agency (EPA) or National Institute for Occupational Safety & Health (NIOSH)] where these apply, and in other cases uses other approved methods (ASTM or in-house). Such standard analytical methods (SAMs) are supplemented by Standard Operating Procedures (SOPs) which provide guidance for a wide range of laboratory activities that do not produce quantitative results, such as calibrating equipment, recordkeeping , formatting procedures, or labeling reagent bottles. Procedures developed in-house are issued through the site's formal procedure review and issue process.

In addition to formal procedures, the laboratory also employs Operator Aids to provide supplemental instructions to certain formal procedures or to provide guidance and clarification for relatively simple tasks where formal procedures are not warranted. Operator Aids are controlled documents that are issued in accordance with current site procedures.

5.7.3.8 External QC Programs

The Portsmouth Analytical Laboratory participates in a wide variety of external control programs for both radiological and non-radiological parameters. These programs vary from time to time. Some of the programs that PORTS has participated in over the years include: the EPA Discharge Monitoring Report (QA) Study, the NIOSH Proficiency Analytical Testing (PAT) Program, the EPA Water Pollution Performance Evaluation Study (WP), the EPA Water Supply Study (WS), the NIOSH Environmental Lead Proficiency Analytical Testing Program (ELPAT), the Proficiency Environmental Testing (PET) program, a commercial program sponsored by the Analytical Products Department of Belpre, Ohio, the EPA Intercomparison Radionuclide Control Program, administered by the EPA Environmental Monitoring System Laboratory at Las Vegas (EMSL-LV), the DOE Environmental Measurements Laboratory (EML) Radionuclide Quality Assessment Program, and the DOE's Mixed Analyte Performance Evaluation Program (MAPEP).

Parameters analyzed in the EMSL-LV program have included gross alpha and gross beta activity, total uranium, and ^{239}Pu in an aqueous matrix. Various matrix samples such as, water, air filters, soil, tissue, and vegetation have been analyzed for a wide variety of radioactive isotopes as part of the EML program.

The MAPEP Program has tested for a variety of metals and for activity of various radionuclides, including Cs-137, Co-57, Co-60, and Pu-239, in soils and waters.

In addition, the laboratory has taken advantage of opportunities to participate in the DOE's New Brunswick Laboratory Safeguards Measurement Evaluation Program for uranium purity in UF_6 and uranyl nitrate, as well as in the Regular European Interlaboratory Measurement Evaluation Program (REIMEP) for uranium isotopic measurements.

5.7.3.9 Laboratory Certifications and Accreditations

The PORTS Analytical Laboratory is accredited and/or certified for non-NRC regulated analyses by various agencies based on analyses being performed and regulatory requirements. Accreditations and/or certifications involve regular onsite inspections by the concerned agencies and

acceptable performance in designated control programs.

5.7.3.10 Statistical Evaluation and Control Charting

Laboratory statisticians and analysts routinely prepare statistical evaluations of QC data, which form the basis for control charts. The laboratory tracks daily balance calibration verifications, laboratory check standard recoveries, and relative percent difference data, among others, as part of its overall process control program.

5.7.3.11 Surveillance, Problem Reporting, and Corrective Action Tracking

Among the important checks and balances of laboratory performance are the surveillance program, problem reporting and corrective action tracking. Surveillances of laboratory operations (analytical methods, personnel qualification records, waste disposal activities, etc.) are performed by laboratory personnel in compliance with the laboratory QA Plan. Problems noted in surveillances require that a problem report be filed in the plantwide system described in Section 6.9. Corrective actions will then be tracked until completed.

Problem reports may be filed by any member of the laboratory organization for any type of laboratory incident, such as an error in reporting data, unacceptable performance in a QC program, missed regulatory holding times, or failure to follow procedure. Problem reports and associated corrective actions will then be tracked by the plantwide system until completion.

5.7.4 Laboratory Support to Environmental and Waste Radiological Monitoring and Radiological Protection

The laboratory provides support in three general areas to ensure the radiological protection of plant personnel and/or the public: environmental radiological analyses, personnel related radiological analyses and analyses in support of nuclear criticality safety/uranium accountability.

5.7.4.1 Environmental Radiological Analyses

The laboratory performs environmental radiological analyses for the site as well as analyses for uranium by fluorimetry and for fluoride (by ion selective electrode). The data for the latter analyte is included in the tables that follow because of its intimate association with process operations. All samples are taken by site sampling personnel.

Another medium analyzed is alumina, which is the trapping material used in continuous vent samplers. The samplers are fabricated and maintained by laboratory personnel who also collect the samples and submits them to the laboratory for analysis. There are fifteen continuous samplers in operation at the vents listed in Table 5.7-2. The samplers have a common design which embodies criteria in American National Standards Institute (ANSI) standards and includes the vent stack flowmeter, the sample flowmeter, the isokinetic sampling probe, and series collection traps of activated alumina. The samplers operate by withdrawing a measured gas sample flow continuously from the vent stack. Particulate as well as gaseous radioactive and other chemical constituents are extracted from the gas and onto the alumina traps. Uranium isotopes and technetium-99 comprise the radioactive species, while reactive fluorides are the measured chemical species. The primary sampler alumina traps at operating facilities, with the exception of the X-344

Gulper, are changed on a weekly basis or may be changed sooner upon request of the Plant Shift Superintendent if an operational upset occurs. The primary trap at the X-344 Gulper is changed every two weeks because historical weekly data show less than detectable levels of radionuclides. At vents which have not operated since the last primary trap change, the primary traps are not changed. Samplers at operating facilities are inspected per established schedule to assure flow readings are in the normal range and systems are properly functioning. Flowmeters and other critical components are calibrated with NIST traceable standards on a schedule not to exceed 18 months. Sampling inlet lines are wet chemical cleaned annually, and emissions computed from analyses of rinse solutions are included in the vent discharges for the year.

Table 5.7-3 presents a list of environmental matrices, corresponding radiological analytes, and lower limit of detection (LLD) levels for parameters analyzed by counting as examples of services of the laboratory. Table 5.7-4 presents summaries of methods used in environmental radiological monitoring of the media addressed in Table 5.7-3 and also indicates precision and bias values for each analysis. The values presented in these tables are typical of the values obtained using these methods. Precision, bias and LLD values are updated on an ongoing basis as specified by procedure. Alternate analytical methods/techniques maybe used provided comparable reporting limits can be achieved.

5.7.4.2 Personnel-Related Radiological Analyses

The laboratory also performs radiological analyses in support of site personnel radiological safety: bioassay, personal air monitoring and radiation protection wipes. Matrices involved in these programs are urine, air filters, and wipes. For bioassay, routine analyses for uranium are performed by ICP/MS. However, analyses may also be performed by alpha spectroscopy for uranium isotopes when U-234 is requested. Table 5.7-5 lists applicable media, parameters, analytical techniques, and LLDs, as applicable. Analytical method summaries, as well as precision and bias values, are provided in Table 5.7-6. These are, also, updated on an on-going basis as specified by procedure.

5.7.4.3 Nuclear Criticality Safety and Uranium Accountability Support

The laboratory supports nuclear criticality safety (NCS) and accountability via analyses. The laboratory also provides support to uranium accountability by taking samples during the bimonthly cascade inventory. These samples consist of off-stream cells, holding drums and banks, and a profile of the purge cascade. Gas samples are then analyzed for uranium concentration and assay.

Liquid and solid samples from NCS approved containers are analyzed for both uranium accountability and NCS reasons. Purity and assay analysis of UF_6 subsamples of product cylinders, Russian product cylinders, customer supplied enriched feed cylinders, a statistically determined amount of normal and Paducah Product feed cylinders, and tails cylinders are performed for accountability and specification confirmation reasons.

A satellite laboratory in the X-705 Decontamination Facility provides laboratory support to all operations in the X-705

Facility as required. Primary activities are uranium concentration and assay determinations for NCS and handling decisions for decontamination solutions, recovery operations, microfiltration, blending, ion exchange, sump water, and miscellaneous uranium bearing solutions for other plantsite facilities.

The laboratory provides isotopic analysis on gas U-tube samples taken from the cascade withdrawal points on a pre-scheduled basis. These analyses are performed on a 24-hours/day, 7-day/week schedule in order to verify the weight percent ^{235}U withdrawn into parent UF_6 cylinders. The laboratory also provides support for accountability by performing isotopic analysis bi-monthly on gas samples withdrawn from selected cascade locations. These results are used in determining and reporting cascade uranium inventory.

5.7.4.4 Waste Sample Radiological Analyses

The laboratory performs analyses on a variety of samples in support of waste characterization efforts. These samples can be categorized as either liquid or solid wastes. The analyses are performed by proportional counting, liquid scintillation counting, gamma spectroscopy, and alpha spectroscopy. Table 5.7-7 lists applicable media, parameters, analytical technique, and LLD's. Analytical method summaries, as well as precision and bias values, are listed in Table 5.7-8. The precision, bias, and LLDs are updated on an on-going basis as specified by procedure.

5.7.5 Section Deleted

Table 5.7-1 Minimum instrument calibration and verification frequency.

Instrument	Calibration*	Verification
Gas Mass Spectrometer	after maintenance/repair or annually, whichever occurs first	every 10 samples
Gamma Spectrometer	after maintenance/repair or annually, whichever occurs first	daily
Solution Enrichment System	after maintenance/repair or annually, whichever occurs first	daily
Proportional Counters	annually	daily
Liquid Scintillation	daily	daily
ICP/Mass Spectrometer: All except urine	each sample batch	every 10 samples
ICP/Mass Spectrometer: Urine	each sample batch	every 16 samples
Alpha Spectrometer	semiannual	daily
Fluorimeter	each sample batch	every 10 samples
Ion Selective Electrode	varies with sample type	every 10 samples

*NOTE: Calibrations are performed more frequently if indicated by the verification.

Table 5.7-2 Vent Samplers - location, flow ratio and emission limit of detection (grams.)

Monitored Vent*	Flow Ratio**	Uranium	U-235	Tc-99
X-330 CR/BE	3175	<0.159	<0.095	<0.002
X-333 CR	3305	<0.165	<0.099	<0.002
X-333 BE	6044	<0.302	<0.181	<0.003
X-326 SIDE	3872	<0.194	<0.116	<0.002
X-326 TOP	2327	<0.116	<0.070	<0.001
X-326 E JET	7364	<0.368	<0.221	<0.004
X-343 CTO/BE Vent	528	<0.025	<0.015	<0.0002
X-344 CTO/BE Vent	528	<0.025	<0.015	<0.0002
X-344 GULPER	4145	<0.207	<0.124	<0.002
X-333 SE1	957	<0.048	<0.029	<0.0005
X-330 SE2	282	<0.014	<0.008	<0.0001
X-330 SE3	285	<0.014	<0.008	<0.0001
X-326 SE4	285	<0.014	<0.008	<0.0001
X-326 SE5	283	<0.014	<0.008	<0.0001
X-326 SE6	472	<0.024	<0.014	<0.0002

*Explanation of Abbreviations: CR = Cold Recovery; BE = Building Evacuation; SIDE = Side Purge; TOP = Top Purge; SE = Seal Exhaust; CTO - Cold Trap Operations

**Ratio of vent stack flow to sample flow

Table 5.7-3 Environmental Radiological Analyses.

Medium	Parameter	Analytical Technique*	LLD
Biota			
Crops	Tc-99	Liquid Scintillation	3×10^{2} pCi/kg
	Uranium (Total)	Fluorimetry	NA
Vegetation	Tc-99	Liquid Scintillation	3×10^{2} pCi/kg
	Uranium (Total)	Fluorimetry	NA
	Fluoride	Ion Selective Electrode	NA
Animal Tissue	Gross Alpha	Proportional Counting	5×10^{3} pCi/kg
	Gross Beta	Proportional Counting	1.0×10^{4} pCi/kg
	Tc-99	Liquid Scintillation	3×10^{2} pCi/kg
	Uranium (Total)	Fluorimetry	NA
	Fluoride	Ion Selective Electrode	NA
Environmental Waters	Gross Alpha	Proportional Counting	4 pCi/L
	Gross Beta	Proportional Counting	8 pCi/L
	Tc-99	Liquid Scintillation	22 pCi/L
	Uranium (Total U-235 & U-238)	ICP/MS	NA
	Fluoride	Ion Selective Electrode	NA
Soils and Sediments	Gross Alpha	Proportional Counting	5×10^{3} pCi/kg
	Gross Beta	Proportional Counting	1.0×10^{4} pCi/kg
	Tc-99	Liquid Scintillation	2×10^{2} pCi/kg
	Uranium (Total)	Fluorimetry	NA
	Fluoride	Ion Selective Electrode	NA
Continuous Vent Monitors	Tc-99	Liquid Scintillation	1×10^{-3} μg/g alumina
	Uranium (total)	Fluorimetry	NA
	U-235	Gamma Spectroscopy	6×10^{-2} μg/g alumina

* Alternate analytical techniques may be used provided comparable reporting limits can be achieved.

Table 5.7-4 Analytical Methods for Environmental Radiological Monitoring.

BIOTA

Tc-99 (Crops, Vegetation, Animal Tissue)

An aliquot of dried vegetation/animal tissue sample is weighed into a porcelain dish, charred on a hot plate, and then ashed in a muffle furnace. Technetium is leached from the sample with sulfuric acid, using potassium persulfate as an oxidant. After filtering, technetium is extracted into concentrated tributyl phosphate and the Tc-99 activity is determined by liquid scintillation counting.

Precision = 3.78% and Bias = -11.7% at 4.7 pCi/g

Total Uranium (Vegetables and Vegetation)

The sample is oven-dried, ignited in a muffle furnace, and the residue is fused with potassium pyrosulfate. The fused sample is dissolved with water. The sample solution is passed through strongly basic anion exchange resin which absorbs the uranium. The uranium is eluted from the resin with 10 percent nitric acid and analyzed by fluorimetry.

Precision = 8.1% and Bias = -7.7% at 10 μg/g spike

Fluoride in Vegetation, Crops and Animal Tissue

The sample is oven-dried, pulverized in a food processor, and digested in potassium hydroxide. Phosphoric acid is added and the pH of the solution is adjusted with hydrochloric acid. The fluoride is determined with an ion selective electrode using the standard addition technique.

Vegetation Precision = 14.8% and Bias = -3.3% at 25 μg/g spike
Animal Tissue Precision = 11.0% and Bias = -8.4% at 25 μg/g spike

Gross Alpha/Gross Beta (Animal Tissue)

A (previously dried and rolled) sample aliquot is leached with hot nitric acid and peroxide, and a portion of the leachate is counted for gross alpha and/or beta radioactivity in a gas flow proportional counter.

Gross Alpha Precision = 6.43% and Bias = 1.56% at 300 dpm
Gross Beta Precision = 5.39% and Bias = 4.04% at 300 dpm

Total Uranium (Animal Tissue)

Whole ground animal tissue is acid digested in a closed Teflon PFA (fluoropolymer resin) vessel using microwave heating for analysis of uranium by fluorimetry.

Precision = 7.0% and Bias = 0.7% at 50 μg/g spike

Table 5.7-4 (Continued)

Water

Gross Alpha/Gross Beta

An aliquot of the preserved water sample is evaporated to a small volume and transferred quantitatively to a tared 2-inch stainless steel counting planchet. The sample residue is dried to a constant weight, reweighed to determine residue weight, and then counted for alpha and/or beta radioactivity.

Gross Alpha Precision = 4.31% and Bias = -5.99% at 72 dpm
Gross Beta Precision = 5.41% and Bias = -1.74% at 72 dpm

Tc-99

An aliquot of the water sample is made 6N in sulfuric acid and oxidized with potassium permanganate. Technetium is extracted into concentrated tributyl phosphate, and the Tc activity in the extract is determined by liquid scintillation counting.

Precision = 4.82% and Bias = 0.62% at 58 dpm

Uranium by Inductively Coupled Plasma/Mass Spectrometry

An acid-preserved sample is thoroughly mixed, and a measured volume is withdrawn and filtered through a filter syringe. A gold internal standard is added to the water sample. The sample solution is nebulized into the Inductively Coupled Plasma/Mass Spectrometer, which measures the total uranium by a quadrupole mass spectrometer.

Precision = 7.9% and Bias = 0.2% at 4 - 6 ug/L

Fluoride

Total ionic strength adjustment buffer (TISAB) solution is added to an aliquot of sample which is then analyzed with an ion selective electrode (ISE).

Precision = 4.0% and Bias = 2.7% at 1.0 - 2.0 mg/L

Soils and Sediments

Gross Alpha/Gross Beta in Soils

An aliquot of a sample is leached with hot nitric acid and peroxide, and a portion of the leachate is analyzed for gross alpha/and or beta radioactivity.

Gross Alpha Precision = 6.43% and Bias = -1.56% at 300 dpm
Gross Beta Precision = 5.39% and Bias = 4.04% at 300 dpm

Table 5.7-4 (Continued)

Tc-99 in Soils and Sediments

A dried sample is weighed into a porcelain dish and ashed. Technetium is leached from the sample with sulfuric acid using potassium persulfate as an oxidant. After centrifuging, the beta activity is extracted using tributyl phosphate and determined by liquid scintillation counting.

Precision = 5.80% and Bias = -3.16% at 86.1 pCi/g

Total Uranium by Fluorimetry

An aliquot of dried, powdered sample is fused with potassium pyrosulfate to solubilize the uranium. The uranium is dissolved in dilute nitric acid and separated from impurities by solvent extraction using TOPO in cyclohexane. An aliquot of the organic extract is fused in a flux containing a mixture of sodium fluoride and sodium carbonate. The uranium fluorescence of the pellet is measured and compared to the fluorescence of a known uranium standard.

Precision = 8.2% and Bias = 2.3% at 5 - 10 mg/kg

Fluoride

A dried homogenized portion of the soil is fused with a potassium-sodium carbonate flux. The melt is dissolved with water and acidified with phosphoric and hydrochloric acids. The fluoride is determined with an ion selective electrode.

Precision = 15.7% and Bias = 4% at 250 μg/g spike

Table 5.7-4 (Continued)

Continuous Vent Monitors

Tc-99

A sample of trap alumina is leached with nitric acid and made 6N with sulfuric acid in the presence of potassium permanganate. After cooling, the solution is extracted with tributyl phosphate, and Tc-99 activity is determined by liquid scintillation counting.

Precision = 3.98% and Bias = -3.18% at 1,097 dpm

Total Uranium

A sample of trap alumina is leached with nitric acid, and the uranium is extracted with TOPO. Total uranium is then determined by fluorimetry.

Precision = 12.06% and Bias = -7.18% at 20 μgU

U-235

Trap alumina is weighed directly into counting bottles, and the U-235 concentration is determined by gamma spectroscopy.

Precision = 6% and Bias = 4.9% at 0.72 $\mu gU^{235}/g$

Table 5.7-5 Personnel-Related Radiological Analyses.

Medium	Parameter	Analytical Technique*	LLD
Urine	U-238	ICP/MS	NA
	U-235	ICP/MS	NA
	Uranium Isotopes	Alpha Spectroscopy	0.1 pCi/L
	Technetium-99	Liquid Scintillation	10 dpm/mL
	Fluoride	Ion Selective Electrode	NA
IH/HP Filters	Uranium Isotopes	Alpha Spectroscopy	0.1 pCi
	Transuranic Isotopes	Alpha Spectroscopy	0.1 pCi
	Thorium Isotopes	Alpha Spectroscopy	0.1 pCi
	Fluoride	Ion Selective Electrode	
	Gaseous Fluoride		NA
	Particulate Fluoride		NA
HP Wipes	Uranium Isotopes	Alpha Spectroscopy	0.1 pCi
	Transuranic Isotopes	Alpha Spectroscopy	0.1 pCi
	Thorium Isotopes	Alpha Spectroscopy	0.1 pCi

* Alternate analytical techniques may be used provided comparable reporting limits can be achieved.

Table 5.7-6 Analytical Method Summaries for Personnel-Related Radiological Analyses.

Uranium and U Isotopes in Urine by Inductively Coupled Plasma/Mass Spectrometry

A urine sample is digested and wet oxidized with strong nitric and hydrochloric acids to solubilize uranium and to destroy the organic matter. Uranium is selectively separated from the chloride salts by an anion exchange resin and is extracted with dilute nitric acid. The uranium isotopes are measured by inductively coupled plasma-mass spectrometry (ICP/MS).

U-235 Precision = 12% and Bias = 7.4% at 15 ng/L
U-238 Precision = 12% and Bias = -7.3% at 85 ng/L
Total U-235 and U-238 in Urine Precision = 11.1% and Bias = -4.8% at 100 ng/L
Total U-235 and U-238 in Urine Precision = 7.7% and Bias = -2.2% at 400 ng/L

Isotopic Uranium in Urine by Alpha Spectroscopy

The urine is preconcentrated with a calcium phosphate co-precipitation and then wet ashed. The uranium is extracted onto a TRU-spec™ column, and eluted with ammonium oxalate. The uranium is then precipitated as its respective fluoride. The precipitate is mounted on a filter paper and counted in a calibrated alpha spectrometer.

U Precision = 5.4% and Bias = -1% at 6.2 dpm

Tc-99 in Urine

To an aliquot of the sample sulfuric acid and potassium persulfate are added, and the oxidation of the sample is completed by heating. After the sample is cooled, Tc-99 is extracted from the sample using tributyl phosphate. The extract is then mixed with a scintillation cocktail and counted by liquid scintillation.

Precision = 6.4% and Bias 0% at the 49 dpm/mL activity.

Fluoride in Urine by Ion Selective Electrode

A urine sample to which a chelating agent has been added is buffered with total ionic strength activity buffer, and fluoride is determined using a fluoride ion selective electrode (ISE) and a combination pH reference electrode.

Precision = 7.8% and Bias = +1% at 1.0 mg/L

Table 5.7-6 (Continued)

Uranium, Transuranic and Thorium Isotopes in Health Physics Filters and Health Physics Wipes

The filter/wipe is leached in a nitric acid-hydrofluoric acid mixture. The thorium, uranium, and transuranics are then extracted onto a TRU-spec™ column, and the individual elements are sequentially eluted. Each fraction is then precipitated as its respective fluoride. The precipitate is mounted on a filter paper and counted in a calibrated alpha spectrometer.

^{241}Am Precision = 6.7% and Bias = 0% at 0.45 - 9.0 pCi
^{239}Pu Precision = 9.8% and Bias = 3% at 0.54 - 7.1 pCi
^{234}U Precision = 21% and Bias = 8% at 0.22 - 2.7 pCi
^{238}U Precision = 13% and Bias = 2% at 0.23 - 2.5 pCi
^{237}Np *
Th isotopes *
*No reference material available; QA obtained from spiked sample results.

Fluoride in Industrial Hygiene Air Samples by ISE

Samples are collected by passing air through a pretreated prefilter paper and backup pad. To determine particulate fluoride, the filter paper is digested with sodium hydroxide fusion and analyzed by ion selective electrode following pH adjustment and addition of TISAB solution. The backup pad is leached in a deionized water/TISAB solution and analyzed by ion selective electrode to determine gaseous fluoride.

Gaseous Fluoride Precision = 6.0% and Bias = 4.2% at 50 μg/filter
Particulate Fluoride Precision = 6.5% and Bias = 1.5% at 50 μg/filter

Table 5.7-7 Waste Sample Radiological Analyses.

Medium	Parameter	Analytical Technique*	LLD
Liquid Waste	Gross Alpha	Proportional Counting	4 pCi/L
	Gross Beta	Proportional Counting	8 pCi/L
	Tc-99	Liquid Scintillation	22 pCi/L
	U-235	Gamma Spectroscopy	0.6 μg U-235/L
	U Isotopes	Alpha Spectroscopy	0.1 pCi/L
	Th Isotopes	Alpha Spectroscopy	0.1 pCi/L
	Am Isotopes	Alpha Spectroscopy	0.1 pCi/L
	Pu Isotopes	Alpha Spectroscopy	0.1 pCi/L
	Np Isotopes	Alpha Spectroscopy	0.1 pCi/L
Solid Waste	Gross Alpha	Proportional Counting	6 pCi/g
	Gross Beta	Proportional Counting	13 pCi/g
	Tc-99	Liquid Scintillation	0.7 pCi/g
	U-235	Gamma Spectroscopy	0.06 μg U-235/g
	U Isotopes	Alpha Spectroscopy	0.1 pCi/g
	Th Isotopes	Alpha Spectroscopy	0.1 pCi/g
	Am Isotopes	Alpha Spectroscopy	0.1 pCi/g
	Pu Isotopes	Alpha Spectroscopy	0.1 pCi/g
	Np Isotopes	Alpha Spectroscopy	0.1 pCi/g

* Alternate analytical techniques may be used providing comparable reporting limits can be achieved.

Table 5.7-8 Analytical Methods for Waste Sample Radiological Analyses.

Liquid Waste

Gross Alpha/Gross Beta

An aliquot of sample is transferred to a tared 2-inch stainless steel planchet, evaporated to dryness, reweighed to determine residue weight, and counted by proportional counting for alpha and/or beta activity.

Gross Alpha: Precision = 4.31% and Bias = -5.99% at 72 dpm
Gross Beta: Precision = 5.41% and Bias = -1.74% at 72 dpm

^{99}Tc in Aqueous Waste

An aliquot of the sample is made 3N in sulfuric acid and oxidized with potassium permanganate. The technetium is extracted into tributyl phosphate, and the ^{99}Tcs determined by liquid scintillation counting.

Precision = 4.82% and Bias = +0.62% at 58 dpm

^{99}Tc in Organic Waste

Amyl acetate is added to the sample, and the technetium is removed from the organic phase with a sodium hydroxide solution. The aqueous extract is then made 6N in sulfuric acid, oxidized with potassium persulfate, and the technetium is extracted with tributyl phosphate. The ^{99}Tc beta activity is then measured by liquid scintillation counting.

Precision = 7.18% and Bias = -1.71% at 1,097 dpm

^{235}U by Gamma Spectroscopy

The sample is transferred into a 125 mL polyethylene bottle and counted for ^{235}U activity in a calibrated high-purity germanium detector.

Precision and Bias to be Established

Table 5.7-8 (Continued)

U, Th, Np, Am, Pu Isotopes by Alpha Spectroscopy

An aliquot of the sample is made 3M in nitric acid and the actinides are extracted with a TRU-Spec™ column. The actinides are sequentially eluted, precipitated as their respective fluorides, filtered, and counted by alpha spectroscopy.

Am: Precision = 7.4% and Bias = +1.0% at 1.9 pCi/L
Pu: Precision = 8.8% and Bias = -4.6% at 2.5 pCi/L
U: Precision = 9.8% and Bias = +12% at 1.1 pCi/L
Np: Precision and Bias to be Established
Th: Precision and Bias to be Established

Solid Waste

Gross Alpha/Gross Beta

The sample is leached in nitric acid and an aliquot of the leachate is evaporated to dryness on a tared 2-inch stainless steel planchet. The planchet is weighed to determine residue weight, and then counted for alpha and/or beta activity by proportional counting.

Gross Alpha: Precision = 6.73% and Bias = -5.33% at 1500 dpm
Gross Beta: Precision = 6.90% and Bias = +2.97% at 1500 dpm

^{99}Tc

The sample is leached with sulfuric acid, oxidized with potassium permanganate, and the technetium is extracted with tributyl phosphate. The ^{99}Tc beta activity is then determined by liquid scintillation counting.

Precision = 5.37% and Bias = +1.45% at 1097 dpm

^{235}U by Gamma Spectroscopy

The sample is leached with nitric acid and the leachate is transferred to a 125 mL polyethylene bottle. The bottle is transferred to a calibrated high-purity germanium detector and counted for ^{235}U activity.

Precision and Bias to be Established

Table 5.7-8 (Continued)

U, Th, Np, Am, Pu Isotopes by Alpha Spectroscopy

The sample is leached in nitric acid and the actinides are selectively extracted with a TRU-Spec™ column. The actinides are then sequentially eluted, precipitated as their respective fluorides, filtered, and counted by alpha spectroscopy.

Am:	Precision = 12% and Bias = -5.8% at 0.6 pCi/g
Pu:	Precision = 7.1% and Bias = -2.9% at 0.6 pCi/g
U:	Precision and Bias to be Established
Np:	Precision = 11% and Bias = -2.5% at 0.6 pCi/g
Th:	Precision and Bias to be Established

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6. ORGANIZATION AND OPERATING PROGRAMS

6.1 ORGANIZATION AND RESPONSIBILITY

Under the Energy Policy Act of 1992, the Nuclear Regulatory Commission (NRC) was required to establish a certification process to ensure that USEC complies with the standards established by NRC for the GDPs under 10 CFR Part 76. The Energy Policy Act provides that the requirement for a certificate of compliance shall be in lieu of any requirement for a license for the GDPs and that USEC shall apply for a certificate of compliance. The Act also requires that USEC and any contractor operating the GDPs for USEC provide the NRC with ready access to the facilities, personnel, and information that the NRC considers necessary to carry out its responsibilities for certification.

USEC is committed to conducting operations in a manner that protects the health and safety of workers and the public, protects the environment and provides for the common defense and security. In order to meet these commitments, as well as others required for operation of the uranium enrichment enterprise, USEC has issued an operations policy manual which contains statements of policy and procedures to guide the day-to-day business and provide direction to USEC employees. The USEC policy with respect to nuclear safety, safeguards and security states in part:

> USEC is ultimately responsible for, and is committed to, safe operation, maintenance, modification, design, fabrication, and testing of the Portsmouth Gaseous Diffusion Plant...and the Paducah Gaseous Diffusion Plant...and is committed to conducting operations in a manner that protects the health and safety of workers and the public, protects the environment and provides for the common defense and security...USEC has provided the management structure to ensure that the safety/safeguards policy is effectively implemented.

The operations organization is responsible for the safe operation of the GDP. Programs and staff organizations are established for safety, safeguards, quality, environmental and health areas and are provided with sufficient resources to support safe operation of the GDP.

USEC has established management systems with associated policies, administrative procedures, and management controls to ensure: the GDP equipment, facilities and procedures; the staff (including training and qualifications), and the programs provide for the protection of the health and safety of workers and the public, protection of the environment, and for the common defense and security. Management controls have been established to maintain the safety/safeguards basis of the GDP as described in this application. Organizations with environmental, health, safety, safeguards and quality responsibilities have been established with a reporting chain, independent from the operations organizations, to a senior site or headquarters manager

(Vice President, Operations, for Corporate Headquarters or General Manager for plant organizations).

The integration of the plant operations and the various programs and requirements is currently accomplished through a variety of management practices. Where frequencies are identified below, they represent current normal business practice but can be adjusted by the manager responsible for the meeting.

- Vice President, Operations, periodic officers and staff meetings, phone call with General Managers to discuss the overall status of the plants, problem and event reports, the daily production report, future plans, and other pertinent subjects.
- General Manager meets weekly with Transfer and Shipping Plant Manager and Organization Managers to discuss issues and policy implementation
- Monthly review of plant performance indicators.
- The Plant review and approval systems [particularly the function of the Plant Operations Review Committee (PORC)] for procedures, training, modifications/changes to the physical plant and/or plant programs as described in this application, provide for the integration of the various requirements and controls.
- Plant work permit systems provide the integration in the field of various health, safety, and environmental program requirements and hazard evaluations.

Figure 6.1-1 shows the USEC organization for operation of the GDPs. Managerial positions that have responsibilities important to safety and safeguards are described in this chapter; other managerial positions are described in those portions of the application where applicable, particularly in the Quality Assurance Program Description, Radioactive Waste Management Program Description, Emergency Plan, Fundamental Nuclear Material Control Plan, Security Plan for the Transportation of Special Nuclear Material of Low Strategic Significance, and the Security Plan for Classified Matter. "Organization Managers" report to either the Transfer and Shipping Plant Manager or General Manager; "Group Managers" report to Organization Managers; and "Section Managers" report to Group Managers.

Section 6.1.1 describes the organizational commitments, relationships, responsibilities and authorities for the overall management system to assure the protection of the health and safety of the workers and the public, protection of the environment, and provide for the common defense. This section includes the qualifications, functions, responsibilities, and authorities of the positions in the organizations assigned functions related to environmental, health, safety, safeguards and quality.

Section 6.1.2 describes the management controls for maintaining the environmental, health, safety, safeguards and quality programs and the administrative systems to control relationships and interfaces between programs.

6.1.1 Organizational Commitments, Relationships, Responsibilities, and Authorities

USEC is committed to the safe operation of the GDP and has provided the management structure to ensure that the safety/safeguards policy is effectively implemented. The management structure provides for line responsibility for safe operations with sufficient staff support to develop, communicate, and provide technical programs for various environmental, health, safety, safeguards and quality areas. The organization of various support staff are provided in the description of the environmental, health, safety, safeguards and quality areas.

USEC provides direction and management of GDP operations. Policy and program direction is provided through the Director, Nuclear Regulatory Affairs, and the Director, Environmental Affairs for their respective areas. The Production Support Manager, who provides direction for the site environmental, safety and health programs, has direct access to the General Manager for matters relating to those programs. These individuals are independent from day-to-day production, plant operating cost, and production scheduling considerations. Also, the Nuclear Safety and Quality Manager (located onsite) who reports to the Vice President, Operations, provides oversight and assurance that corporate policies and procedures are being followed in operation of the plant.

The General Manager directs and oversees site activities to ensure safe, reliable, and efficient operations. The Transfer and Shipping Plant Manager reports to the General Manager and directs and coordinates the production plant operation in accordance with policies as reflected in plant procedures and practices. The production line organizations report to the Transfer and Shipping Plant Manager and have responsibilities for implementation of safety and safeguards policies and procedures in daily operations. The on-duty PSS reports to the Shift Operations Manager and provides direction and coordination for shift operations. The Shift Operations Manager reports to the Operations Manager. The staff and support organizations report to the General Manager and provide program direction and support to the production line in implementing safety and safeguards requirements. Finally, administrative organizations report to the General Manager and provide the services required to support the overall plant operations.

Personnel minimum qualifications, functions and responsibilities for key staff positions are described below. Throughout this section, equivalent technical experience means the substitution of 2 years of nuclear industry experience for each year of college up to a total of 3 years. Additionally, 30 semester hours or 45 quarter hours from an accredited college or university may be substituted for the remaining 1 year of baccalaureate education. Individuals who do not possess the formal educational requirements specified in this section or do not meet the equivalent technical experience defined above shall not be automatically eliminated where other factors provide sufficient demonstration of their abilities to fulfill the duties of a specific position. These other factors must clearly demonstrate proficiency in the technical area for which the position will be responsible, for example, a license or certification, documented completion of relevant training, or previous experience in the same position at another facility. These factors shall be evaluated on a case-by-case basis, documented, and approved by the General Manager or appropriate Headquarters management, with the consultation of the human resources organization.

6.1.1.1 Section Deleted

6.1.1.2 Vice President, Operations

The Vice President, Operations, reports to the Executive Vice President and Chief Operating Officer (EVP-COO).

The Vice President, Operations has overall responsibility for safe operations of the GDPs, including packaging and transportation of radioactive material. The Vice President, Operations is authorized to direct the General Manager to take any specific action, including but not limited to, placing all or any portion of one or both GDPs in a safe condition, in order to ensure health and safety of workers and the public, protection of the environment, safeguards and security, and to achieve or maintain compliance with applicable regulatory requirements. In addition, the Vice President, Operations must concur with the decision of the General Manager to restart any operation that was directed to be shut down by the Vice President, Operations or by the Nuclear Safety and Quality Manager.

The Vice President, Operations, has implementation responsibility for all activities within the GDP organizations, including the functions of operations, maintenance, plant support, engineering, transportation, procurement, materials handling and storage, and industrial, radiological, and nuclear safety.

The Vice President, Operations is responsible for both the Quality Assurance Program (QAP) and the Packaging and Transportation Quality Assurance Program (PTQAP), and for determining the status, adequacy, and effectiveness of the QAP as described in section 2.2.5 of the QAP and the PTQAP.

The Vice President, Operations, has implementation responsibility for packaging and transportation activities within USEC. The Vice President, Operations, is responsible for final approval of the design of packaging and design changes to packaging used for radioactive material shipments.

The Vice President, Operations, shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, 6 years nuclear experience, and 10 years management experience (which may be concurrent with the nuclear experience).

The Vice President, Operations, is appointed by the USEC Board of Directors.

6.1.1.3 Director, Nuclear Regulatory Affairs

The Director, Nuclear Regulatory Affairs, located at headquarters, reports to the Vice President, Operations.

This position has responsibility for the management of nuclear regulatory functions and the corporate policy system. This individual is the primary day-to-day interface with the NRC Office of Nuclear Materials Safety and Safeguards (NMSS) and has overall responsibility for management of activities related to certification of the GDPs. Although this individual works closely with the General Manager and key plant personnel, he/she is independent from production, plant operating cost, and production schedule concerns, and has the authority to stop work if there is a failure to adhere to regulatory requirements within his/her area of responsibility. The Plant Nuclear Regulatory Affairs Manager reports to the General Manager, but is governed by and must adhere to policies established by the Director, Nuclear Regulatory Affairs.

This position shall have as a minimum a bachelors degree or equivalent technical experience, and 4 years of nuclear experience. This individual is appointed by the Vice President, Operations.

6.1.1.4 Director, Engineering

The Director, Engineering, reports to the Vice President, Operations.

This position is responsible for the management of USEC engineering functions to ensure that design bases and safety bases are maintained and that plant and company objectives are attained. This position ensures that engineering services within USEC support operations and maintenance activities to meet safety requirements, production goals, and budgets. The Director, Engineering, has stop work authority for any activity that would be or is in violation of the plant safety basis, the Technical Safety Requirements, or the requirements and assumptions of the accident analyses.

This position shall have as a minimum, a bachelor's degree in engineering or the physical sciences and four years of nuclear experience with at least six months in a gaseous diffusion plant. This individual is appointed by the Vice President, Operations.

6.1.1.5 Director, Environmental Affairs

The Director, Environmental Affairs, located at headquarters, reports to the Vice President, Operations.

This individual is responsible for the development and management of corporation-wide environmental and waste management policies and for these activities at the plants. This individual is independent from production, plant operating costs, and production schedule concerns, and has the authority to stop work if there is a failure to adhere to applicable regulatory requirements in environmental and waste management areas.

This position shall have as a minimum a bachelors degree or equivalent technical experience, and 4 years of environmental management experience. This individual is appointed by the Vice President, Operations.

6.1.1.6 Nuclear Safety and Quality Manager

The Nuclear Safety and Quality Manager, located at the plant, reports to the Vice President, Operations.

The Nuclear Safety and Quality Manager has the responsibility to exercise 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 policies, procedures and management expectations. The Nuclear Safety and Quality Manager is also responsible for nuclear material control and accountability, and quality control, including receipt inspection of material and inspection of other selected safety-related activities.

The Nuclear Safety and Quality Manager is independent from production, plant operating cost, and production schedule concerns, and has authority to shut down operations and/or stop work when necessary to ensure protection of public and worker health and safety and provide for common defense and security and to ensure regulatory and quality compliance. This manager has access to all information at the site related to safety, safeguards and quality. This manager interacts directly with the General Manager, other managers, and key plant personnel and participates, as desired, in any evaluations or discussions related to safety, safeguards and quality. This manager informs the Vice President, Operations, and the General Manager about safety, safeguards, and quality issues and compliance. This manager manages the Nuclear Safety and Quality Office and directs plant quality assurance functions involving audits and oversight of plant operations as well as a nuclear safety assurance function.

The Nuclear Safety and Quality Manager primarily works the office shift but periodically observes plant operations performed on backshifts. This manager is on-call and is notified by the Plant Shift Superintendent's office of reportable events, per plant procedures. This notification occurs on all shifts.

The Nuclear Safety and Quality Manager shall have as a minimum a technical degree and 15 years nuclear experience with 3 years of management experience in quality assurance, nuclear safety oversight, engineering and technical support, or regulatory affairs. Either the Nuclear Safety and Quality Manager, or a management position responsible for quality assurance that reports to the Nuclear Safety and Quality Manager, shall have a minimum of 1 year quality assurance experience or 1 year experience implementing quality assurance program requirements.

The Nuclear Safety and Quality Manager is appointed by the Vice President, Operations.

6.1.1.7 USEC NCS Manager

The USEC NCS Manager reports to the Director, Engineering.

The position is responsible for the management of USEC nuclear criticality safety functions, including developing and implementing the nuclear criticality safety programs at both sites, and conducting assessments of program implementation. These duties include programmatic oversight of nuclear criticality safety and nuclear criticality safety training. The USEC NCS Manager shall have direct access to the Director, Engineering, and to the General Managers at both sites for issues involving nuclear criticality safety matters or concerns, and has stop work authority for any activity at either site that could cause a nuclear criticality safety concern.

The USEC NCS Manager shall have as a minimum a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years nuclear experience, including 6 months at a uranium processing facility where nuclear criticality safety was practiced.

The USEC NCS Manager is appointed by the Director, Engineering.

6.1.1.8 General Manager

The General Manager reports to the Vice President, Operations.

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 policies. The General Manager is responsible for production, training, procedures, plant services, engineering, transportation, materials handling and storage, and occupational, environmental and nuclear safety. Day-to-day authority and accountability for production and production support activities is assigned to the Transfer and Shipping Plant Manager. The General Manager has responsibility for the primary day-to-day interface with NRC on matters of adequate safety/safeguards and regulatory compliance, and may delegate responsibility for that interface to the Nuclear Regulatory Affairs Manager.

The General Manager has shut down and stop work authority for all or any portion of the plant (leased facilities). The General Manager shall be responsible to authorize restart of shut down operations and must obtain concurrence of the Vice President, Operations, for any operations that were directed to be shut down by the Vice President, Operations, or by the Nuclear Safety and Quality Manager.

The General Manager shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, 6 years of nuclear experience, and 6 years of management experience (which may be concurrent with the nuclear experience).

The General Manager is appointed by the Vice President, Operations.

6.1.1.9 Transfer and Shipping Plant Manager

The Transfer and Shipping Plant Manager reports to the General Manager.

The Transfer and Shipping Plant Manager is responsible for the day-to-day production activities at the site including operations, maintenance, and production support. The Transfer and Shipping Plant Manager shall be responsible for authorization of restart of shutdown operations but must seek concurrence from the General Manager for any operation that was shutdown by the General Manager, the Vice President, Operations, or the Nuclear Safety and Quality Manager.

The Transfer and Shipping Plant Manager shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, 6 years of nuclear experience, and 6 years of management experience (which may be concurrent with the nuclear experience). The Transfer and Shipping Plant Manager is appointed by the General Manager with concurrences by the Vice President, Operations.

6.1.1.10 Operations Manager

The Operations Manager reports to the Transfer and Shipping Plant Manager.

The Operations Manager is responsible for the operations of the enrichment cascade, plant utilities, chemical services, feed and product facilities and shift operations. This includes activities such as ensuring the correct and safe operation of the UF_6 processes; proper receipt, storage, handling and onsite transportation of UF_6; providing electric power, steam, compressed air, nitrogen, plant and sanitary water, and waste water treatment for the cascade and support facilities; and providing chemical cleaning and decontamination services. The Operations Manager is also responsible for integrated plant scheduling. This includes managing daily work control activities, developing an integrated work schedule, and coordinating development of work control guidelines. In the absence of the General Manager and Transfer and Shipping Plant Manager, the Operations Manager may be delegated the responsibilities and authorities of the General Manager and/or the Transfer and Shipping Plant Manager. This manager shall have the authority to stop work and/or shut down operations in any part of the operation for which he/she has responsibility.

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

The Operations Manager is appointed by the Transfer and Shipping Plant Manager with concurrence by the General Manager and the Vice President, Operations.

6.1.1.11 Maintenance Manager

The Maintenance Manager reports to the Transfer and Shipping Plant Manager.

The Maintenance Manager is responsible for safe and reliable performance of preventive, predictive, and corrective maintenance and support services on plant facilities and equipment. This includes troubleshooting, maintenance of logs and records, work planning interfacing with work control to initiate, screen,

evaluate, and prioritize maintenance work, and coordinating shop maintenance. The manager shall have the authority to stop work and/or shut down operations in any part of the operation for which he/she has responsibility.

The Maintenance Manager shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience with at least 6 months in a gaseous diffusion plant.

The Maintenance Manager is appointed by the Transfer and Shipping Plant Manager with concurrence by the General Manager and by the Vice President, Operations.

6.1.1.12 Production Support Manager

The Production Support Manager reports to the Transfer and Shipping Plant Manager. The Production Support Manager is governed by and must adhere to policies established by the Director, Environmental Affairs.

The Production Support Manager is responsible for oversight functions in environmental, safety and health areas as well as the technical functions in direct support of production activities. These include the health physics/industrial hygiene program, laboratory operations, waste management and environmental compliance, and industrial safety. The Production Support Manager is the senior site manager responsible for environmental, safety and health matters and is responsible for establishing and implementing the environmental monitoring program described in Section 5.1, the site environmental protection programs, the industrial and chemical safety programs, and the health physics program at the facility. This manager appoints a site program manager for the Chemical Safety Program. This manager shall have the authority to stop work and/or shutdown operations in any part of the operation for which he/she has responsibility.

The Production Support Manager shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, and 4 years of nuclear experience with at least 6 months in a gaseous diffusion plant.

The Production Support Manager is appointed by the Transfer and Shipping Plant Manager with concurrence by the General Manager, the Vice President, Operations, and the Director, Environmental Affairs.

6.1.1.13 Radiation Protection Manager

The Radiation Protection Manager reports to the Production Support Manager.

The Radiation Protection Manager is responsible for the implementation, maintenance, and effectiveness of the health physics, radiation protection and industrial hygiene programs. These duties include training personnel in the use of radiological program support equipment, controlling radiation exposure of personnel, determining the radiological status of the facility, determining the need for issuing and closing out radiation work permits, and conducting the radiological occupational monitoring program. The Radiation Protection Manager has direct access to the General Manager and the Transfer and Shipping Plant Manager concerning radiation protection matters and has stop work authority for activities not being conducted in accordance with radiation protection requirements and policies.

The Radiation Protection Manager shall have as a minimum a bachelors degree in engineering, health physics, radiation protection, or the physical sciences or equivalent technical experience, and 4 years experience in radiation protection including 6 months at a uranium processing facility.

The Radiation Protection Manager is appointed by the Production Support Manager with concurrence by the Transfer and Shipping Plant Manager and General Manager.

6.1.1.14 Section Deleted

6.1.1.15 Shift Operations Manager

The Shift Operations Manager reports to the Operations Manager.

The Shift Operations manager coordinates the activities of the Plant Shift Superintendents and provides technical and administrative support.

The Shift Operations Manager shall have as a minimum a bachelors degree or equivalent technical experience, and 4 years nuclear experience with at least 6 months at a GDP.

The Shift Operations Manager is appointed by the Operations Manager with concurrence by the Transfer and Shipping Plant Manager, General Manager and the Vice President, Operations.

6.1.1.16 Plant Shift Superintendent

The Plant Shift Superintendent reports to the Shift Operations Manager.

As the senior manager on shift, 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. The Plant Shift Superintendent is responsible for accumulation and dissemination of information regarding plant activities, serving as or designating an incident commander during plant emergencies, and making notification of events.

The PSS is authorized to stop operations when system operability or the overall safety of operations is in question. The PSS is also authorized to initiate restart after shut down for non-routine reasons. For shutdowns that are directed by the Vice President, Operations; Nuclear Safety and Quality Manager; the General Manager, or the Transfer and Shipping Plant

Manager; the PSS may authorize restart only after obtaining the approval of the Transfer and Shipping Plant Manager (who will in turn obtain the necessary concurrence as described in Section 6.1.1.9).

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

The Plant Shift Superintendent is appointed by the Shift Operations Manager with concurrence by the Transfer and Shipping Plant Manager and General Manager.

6.1.1.17 Nuclear Criticality Safety Manager

The Nuclear Criticality Safety Manager reports to the USEC NCS Manager.

The Nuclear Criticality Safety Manager is responsible for implementing the nuclear criticality safety program for the facility. These duties include technical oversight of nuclear criticality safety; nuclear criticality safety training; the evaluation and approval of current and proposed changes to process conditions, equipment, and procedures involving fissile material operations; and conducting assessments of program implementation. The Nuclear Criticality Safety Manager has direct access to the General Manager concerning nuclear criticality safety matters and has stop work authority for any activity that could cause a criticality concern.

The Nuclear Criticality Safety Manager shall have as a minimum a bachelors degree in engineering or physical sciences, and four years nuclear criticality experience or nuclear engineering experience (e.g., core load design, fuel design, reactor engineering) with at least six months at a uranium processing facility where nuclear criticality safety was practiced.

The Nuclear Criticality Safety Manager is appointed by the USEC NCS Manager with the concurrence of the Director, Engineering and the General Manager.

6.1.1.18 Engineering Manager

The Engineering Manager reports to the Director, Engineering.

The Engineering Manager is responsible for engineering activities in support of operations, including project management, design, fabrication, and construction of plant modifications or additions; systems and reliability engineering; nuclear safety; and configuration management program. The Engineering Manager has stop work authority for any activity that would be or is in violation of the plant safety basis, the Technical Safety Requirements, or the requirements and assumptions of the accident analyses.

The Engineering Manager is the design authority for radioactive material packaging and transportation structures, systems, and components. As such, the Engineering Manager is responsible for the following:

1. Evaluation of supplier's technical capabilities and approval of technical dispositions, and technical evaluation of supplier-generated nonconforming material, equipment, or services.

2. Providing measures which ensure the proper selection, application, methods of acceptance, and use of commercial grade items when applicable.

3. Specifying requirements for handling, storage, shipping, cleaning, packaging and on-site movement of SSC items in specifications, drawings, instruction, procedures, procurement documents, and/or other appropriate documents.

4. Determining applicable special processes, providing technical requirements, review and concurrence for special process procedures including the utilization and application of nondestructive examination procedures.

5. Providing technical criteria for testing and the evaluation, and resolution of test deficiencies.

6. Providing documented technical justification for nonconforming items dispositioned "use-as-is" or "repair" and ensuring as-built records reflect accepted deviations as required.

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

The Engineering Manager is appointed by the Director, Engineering with concurrence by the General Manager and the Vice President, Operations.

6.1.1.19 Nuclear Safety Manager

The Nuclear Safety Manager reports to the Design Control Manager.

The Nuclear Safety Manager is, among other assigned duties, responsible for developing and implementing the safety analysis program for the facility. These duties include technical oversight of safety analysis; safety analysis training; procedures involving fissile material operations; implementation of the unreviewed safety question determination programs; and conducting assessments of program implementation. The Nuclear Safety Manager also is responsible for procurement engineering and configuration management. The Nuclear Safety Manager has direct access to the General Manager concerning nuclear safety matters and has stop work authority for any activity that would be or is in violation of the plant safety basis, the Technical Safety Requirements, or the requirements and assumptions of the accident analysis.

The Nuclear Safety Manager shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, and 4 years nuclear safety experience including 6 months at a gaseous diffusion plant.

The Nuclear Safety Manager is appointed by the Engineering Manager with concurrence by the Director, Engineering and the General Manager.

6.1.1.20 Design Control Manager

The Design Control Manager reports to the Engineering Manager.

The Design Control Manager is responsible for planning, directing, and controlling activities to provide design engineering and control support to the plant.

The position requires a minimum of a bachelors degree in engineering or a physical science, and four years Nuclear Design experience, including six months GDP experience.

The Design Control Manager is appointed by the Engineering Manager, with the concurrence of the Director, Engineering.

6.1.1.21 Plant Services Manager

The Plant Services Manager reports to the General Manager.

The Plant Services Manager is responsible for plant fire, and police services, security, Emergency Management, medical support, records management, and document control. The Plant Services Manager has stop work authority for activities not being conducted in accordance with applicable regulatory requirements.

The Plant Services Manager shall have as a minimum a bachelors degree or equivalent technical experience, and 4 years of nuclear experience with at least 6 months in a gaseous diffusion plant.

The Plant Services Manager is appointed by the General Manager with concurrence by the Vice President, Operations.

6.1.1.22 Security Manager

The Security (Group) Manager reports to the Plant Services Manager.

The Security Manager is responsible for plant police services and security. The Security Manager has direct access to the General Manager concerning security matters and has stop work authority for activities not being conducted in accordance with applicable security requirements.

The Security Manager shall have as a minimum a bachelors degree or equivalent technical experience, and 4 years of security experience.

The Security Manager is appointed by the Plant Services Manager with concurrence by the General Manager.

6.1.1.23 Fire Services Manager

The Fire Services Manager reports to the Plant Services Manager and is governed by, and must adhere to, policies established by the General Manager.

The Fire Services Manager is responsible for plant fire services including interpretation and application of applicable fire codes and standards, and has stop work authority for activities not being conducted in accordance with applicable fire protection requirements. The Fire Services Manager is the senior site fire protection officer.

The Fire Services Manager shall have as a minimum a bachelors degree or equivalent technical experience, 4 years of fire protection experience, and 6 months of nuclear experience.

The Fire Services Manager is appointed by the Plant Services Manager with concurrence by the General Manager.

6.1.1.24 Training Manager

The Training Manager reports to the General Manager.

The Training Manager is responsible for preparation, presentation, and recording of employee orientations, and for technical and qualification training program development and implementation. The Training Manager is also responsible for the development and implementation of the procedures program.

The Training Manager shall have as a minimum a bachelors degree or equivalent technical experience, and 4 years of nuclear experience.

The Training Manager is appointed by the General Manager with concurrence by the Vice President, Operations.

6.1.1.25 Nuclear Regulatory Affairs Manager

The Nuclear Regulatory Affairs Manager reports to the General Manager. The Nuclear Regulatory Affairs Manager is governed by and must adhere to policies established by the Director, Nuclear Regulatory Affairs.

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As delegated by the General Manager, the Nuclear Regulatory Affairs Manager is responsible for the day-to-day interface with NRC representatives on matters of regulatory compliance, event investigation and reporting, and NRC regulatory commitment management. As delegated by the Director, Nuclear Regulatory Affairs, the Nuclear Regulatory Affairs Manager has responsibility for coordinating certification related and certificate renewal-related activities. The Nuclear Regulatory Affairs Manager is also responsible for providing plant management with data to assure the plant's corrective actions and commitments are properly addressed and managed to facilitate compliance with implementing policies and procedures, for the Operating Experience Review Program, for administration of the problem reporting systems, and for alerting plant management to adverse trends as noted for identified deficiencies. The Nuclear Regulatory Affairs Manager shall have as a minimum a bachelors degree in engineering or the physical sciences or equivalent technical experience, and 4 years nuclear experience.

The Nuclear Regulatory Affairs Manager is appointed by the General Manager with concurrence of the Director, Nuclear Regulatory Affairs.

6.1.1.26 Operations and Maintenance First-Line Managers

First-line operations and maintenance managers report to Group or Section Managers.

First-line operations managers shall have as a minimum a high school diploma or satisfactory completion of the General Educational Development (GED) test and 3 years of plant operations experience with at least 6 months in a gaseous diffusion plant. First-line maintenance managers shall have as a minimum a high school diploma or satisfactory completion of the General Educational Development (GED) test and 3 years of maintenance related plant experience with at least 3 months in a gaseous diffusion plant, or 11 years maintenance experience in a gaseous diffusion plant. As an equivalent alternative, a first-line operations or maintenance manager may have a four year technical degree with at least two years gaseous diffusion plant experience in an operational, maintenance, or engineering assignment.

First-line operations and maintenance managers are appointed by Group or Section Managers with concurrence by the Organization Managers.

First-line operations managers can authorize the restart of equipment that has been shutdown in a routine fashion when the prerequisites and limitations of the associated operating procedure are met.

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6.1.1.29 GDP Procurement and Materials Manager

The GDP Procurement and Materials Manager is the senior site manager responsible for receipt, delivery, storage, shipment, control and on-site movement of packaging and transportation SSCs, packaging and transportation activities in support of operations and hazardous chemicals under his cognizance to the point of issuance. This manager interacts directly with the General Manager, other managers and key plant personnel and participates, as desired, in any discussions related to procurement and materials management.

The GDP Procurement and Materials Manager is appointed by and reports to the USEC Director of Procurement and Materials.

6.1.2 Management Controls

The management controls established by USEC include policies and directives, management systems, and administrative procedures. USEC establishes policies which are communicated to the plants. Policies related to the protection of health and safety of workers and the public, protection of the environment, and providing for the common defense are discussed in pertinent sections of this application.

Management systems and programs are described in Chapters 5 and 6, the Technical Safety Requirements, and in several program plans and descriptions (notably, the Emergency Plan, Quality Assurance Program, Radioactive Waste Management Program, Fundamental Nuclear Material Control Plan, and Security Plans). The commitment tracking and corrective action management systems are integrated to prioritize plant actions consistent with their safety and safeguards significance. Where safety or safeguards might be adversely impacted by cost or schedule considerations, it is the policy of USEC to subordinate cost and schedular considerations to ensure adequate treatment of safety and safeguards. The interface and coordination of the elements of safety and safeguards programs are accomplished through the management systems described in this chapter and are manifest in the line direction of plant operations. Additionally, oversight of the integration of various program elements is provided by the Nuclear Safety and Quality Manager. From a technical perspective, the Plant Operations Review Committee (PORC) and its associated subcommittees provide the mechanism for evaluating and integrating safety and safeguards program elements from a plant design and program change perspective.

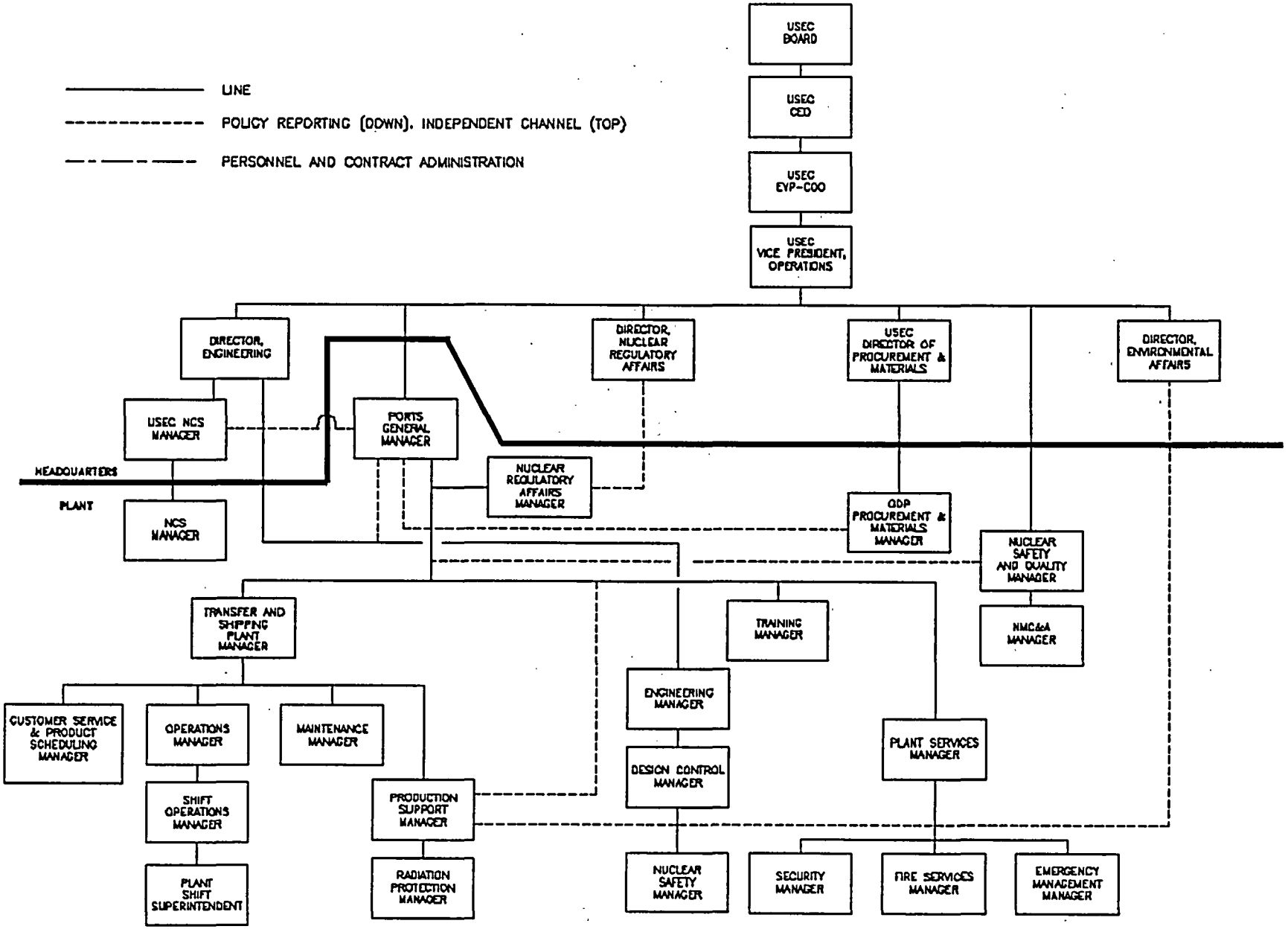

Figure 6.1-1 Uranium Enrichment Facilities Organization Chart

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6.2 SAFETY COMMITTEES

The United States Enrichment Corporation (USEC) has established a Plant Operations Review Committee (PORC) that provides the review of the gaseous diffusion plants (GDPs) activities as required by 10 CFR 76.68(a). The membership, qualifications, meeting frequency and quorum, functions, responsibilities, and required records are as described in TSR 3.10. The PORC advises the General Manager on matters related to nuclear safety. The General Manager approves the procedure implementing the PORC activities described in TSR 3.10. Auditing and oversight of PORC activities is the responsibility of the Nuclear Safety and Quality Manager.

Subcommittees may be established by the PORC chair to provide assistance in conducting reviews and assessments as described in the PORC procedure. The PORC chairperson approves the subcommittee procedures, membership, and member qualifications. The PORC maintains the overall responsibility for all required review responsibilities.

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6.3 PLANT CHANGES AND CONFIGURATION MANAGEMENT

In accordance with 10 CFR 76.68, USEC is permitted to make changes to the plant or to the plant's operations as described in the safety analysis report in accordance with specified requirements. In addition, USEC may apply for amendments to the certificate of compliance in accordance with 10 CFR 76.45, and may modify programs and plans in accordance with the criteria set forth in this application. This section describes the mechanisms utilized to provide for control and approval of such changes. In addition, this section describes the plant systems for controlling the configuration of plant programs and systems in accordance with this certification application, including the Safety Analysis Report (SAR) and Technical Safety Requirements (TSRs).

6.3.1 Technical Safety Requirements and Certificate of Compliance Conditions

Proposed changes to the plant or the plant's operations, including physical modifications, changes to procedures and training, and changes to programs and plans, are reviewed relative to the TSRs and the conditions of the certificate of compliance.

USEC shall not make any changes to the TSRs (except the Basis Statements) or to any conditions of the certificate of compliance, unless prior approval for such change is granted by the NRC in accordance with 10 CFR 76.45. USEC may make changes to the Basis Statements without prior NRC approval. Changes to the Basis statements will be included in revisions to the TSRs and will be submitted annually to the NRC. The revised pages of the TSRs will be marked and dated to indicate each change.

Proposed changes to the TSRs, TSR Basis Statements, and any conditions of the certificate of compliance are authorized by responsible management and approved by the Plant Operations Review Committee (PORC) (see Section 6.2).

Nuclear Regulatory Affairs (NRA) controls documentation and coordinates implementation of approved changes to the TSRs and conditions of the certificate of compliance, controls documentation of changes to TSR Basis Statements, and maintains records of changes and change control documentation.

6.3.2 Safety Analysis Report

Proposed changes to the plant or to the plant's operations, including physical modifications, changes to procedures and training, and changes to programs and plans, are reviewed relative to the description of the plant and plant operations in the SAR.

In accordance with 10 CFR 76.35, the Safety Analysis Report is comprised of Volumes 1 and 2 of the Certification Application and includes:

Chapter 1 - Introduction and General Description of the Facility
Chapter 2 - Site Characteristics of the Portsmouth Gaseous Diffusion Plant
Chapter 3 - Facility and Process Description
Chapter 4 - Accident Analysis
Chapter 5 - Nuclear Safety Programs
Chapter 6 - Organization and Operating Programs

USEC may make changes to the plant or to the plant's operations, as described in the Safety Analysis Report, without prior NRC approval, provided all the following provisions of 10 CFR 76.68(a) are met:

1. USEC shall conduct a written safety analysis which demonstrates that the changes would not result in undue risk to public health and safety, the common defense and security, or to the environment;

2. The changes must be authorized by responsible management and approved by the PORC, as described in Section 6.2;

3. The changes may not decrease effectiveness of the plant's safety, safeguards, and security programs;

4. The changes may not involve a change in any condition to the certificate of compliance;

5. The changes may not involve a change to any condition to the approved compliance plan;

6. The changes may not involve an unreviewed safety question.

If USEC determines that a proposed change would not meet one of the above provisions, USEC shall not make the change unless prior approval for such change is granted by the NRC in accordance with 10 CFR 76.45.

USEC will evaluate any as-found conditions that do not agree with the description of the plant or plant operations as described in the SAR (including programs, plans, policies, and operations) in accordance with the above provisions. Any revisions required to reflect the as-found conditions will be submitted annually as specified in 10 CFR 76.36 or at a shorter interval as may be specified in the certificate.

Changes will be submitted to the NRC annually pursuant to the requirements of 10 CFR 76.36. The revised pages of the SAR will be marked and dated to indicate each such change. NRA is responsible for the maintenance and control of the SAR and maintains records of changes and change control documentation.

6.3.3 Programs and Plans

NRA is responsible for maintaining program plans and documentation of changes as discussed in the following subsections.

6.3.3.1 Programs and Plans Requiring Evaluations

Proposed changes to the Emergency Plan, Quality Assurance Program Description, Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance, Physical Security Plan for the Transportation of Special Nuclear Material of Low Strategic Significance, Security Plan for the Protection of Classified Matter, and the Fundamental Nuclear Material Control Plan shall be evaluated by USEC to determine whether the proposed change would decrease the effectiveness of the program. USEC shall not make a change to these plans or program descriptions that would decrease the effectiveness of the program unless prior approval of such change has been granted by the NRC in accordance with 10 CFR 76.45. These changes shall be authorized by responsible management and approved by the PORC (as described in Section 6.2).

NRA is responsible for maintaining the program plans and documentation of changes. Proposed changes to the plant or to the plant's operations including procedures and training, plant programs, or the plant physical configuration are reviewed relative to the program descriptions and plans.

Those changes to the program descriptions and plans that are determined not to decrease program effectiveness may be made without prior NRC approval with notification of the NRC of changes made as follows:

1. Changes to the Quality Assurance Program (QAP) will be submitted annually;

2. Changes to the Emergency Plan will be submitted to the NRC and to affected offsite organizations within 6 months after the change is made;

3. Changes to the Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance, Physical Security Plan for the Transportation of Special Nuclear Material of Low Strategic Significance, Security Plan for the Protection of Classified Matter, and Fundamental Nuclear Material Control Plan will be submitted to the NRC within 6 months after the change is made;

The revised pages of the program description or plan will be marked and dated to indicate each change.

6.3.3.2 Changes to Other Plans and Information

USEC may change the Radioactive Waste Management Plan, Depleted Uranium Management Plan, Decommissioning Funding Plan, Environmental Compliance Status Report, and Supplemental Environmental Information without prior NRC approval. Changes to these plans and information will be authorized by responsible management and approved by the PORC (as described in Section 6.2). Revisions to these documents reflecting such changes will be submitted annually to the NRC. The revised pages will be marked and dated to indicate each change.

NRA is responsible for maintaining the program plans and documentation of changes. Proposed changes to the plant or to the plant's operations including procedures and training, plant programs, or the plant physical configuration are reviewed relative to the program descriptions and plans.

6.3.4 Records of Plant Changes

Records of changes described in Sections 6.3.1 through 6.3.3, including their safety analysis and documentation, are maintained in accordance with 10 CFR 76.68(c) and as described in Section 6.10.

6.3.5 Physical Plant Change Control and Configuration Management

The control of changes to the physical plant and control of the physical plant configuration are managed by the Design Authority (Engineering). The following paragraphs describe the plant Configuration Management (CM) Program mechanisms utilized to control changes and maintain the plant configuration as described in this certification application, including the SAR and TSRs. The evaluation of changes relative to the application is as described above relative to the particular section of the application involved. As noted, changes affecting the application are incorporated by the NRA organization.

6.3.5.1 Overview

6.3.5.1.1 Objectives

The CM Program has been developed and documented in a site procedure and is implemented to ensure that changes from the plant baseline configuration are controlled to prevent degradation of safety or safeguards. The program includes 1) identification of the structures, systems, equipment, components, and design features credited for safety and safeguards; 2) organizational descriptions of duties and responsibilities; and 3) administrative controls, procedures and policies, to implement and document activities that maintain the plant's baseline for safety and safeguards.

This program includes organizations and administrative processes and assessments to ensure accurate, current design documentation that matches the plant's physical configuration while complying with applicable requirements.

6.3.5.1.2 Safety Benefits

The CM Program results in:

1. Performance of design activities in accordance with the QAP that produces technically sound designs to meet regulatory and industrial code/standards requirements and preserve or increase design margins of safety.

2. Establishment and enforcement of controls that ensure the integrity of the design and design documentation are maintained throughout the plant life, including the development and implementation of modifications.

3. Assurance that coordination of the intent of the design documents with operating and maintenance procedures accompanied by appropriate training of personnel to support plant changes is accomplished.

4. Effective communication of boundaries and systems, structures, and components (SSCs) as described in Section 3.8.

6.3.5.1.3 Program Elements

The elements of the program consist of the following:

1. Program Management,

2. Design Requirements,

3. Records Management and Document Control,

4. Change Control,

5. Assessments, and

6. Training.

6.3.5.1.4 CM Structure

Key responsibilities and functional interfaces are illustrated in Figure 6.3-1.

6.3.5.2 Program Management

6.3.5.2.1 Policy Statement

The configuration of important safety and safeguards features of facilities, activities, processes, and experiments must conform to approved design and administrative requirements. In addition, decisions affecting safety, quality, and other important functions must be documented, and changes must not lead to violation of established requirements.

Important features are structures, systems, equipment components, computer hardware or software, communications networks, instructions or procedures, or other designated physical or administrative items that protect the health and safety of workers and the public; protect the environment; ensure compliance with applicable laws, orders, or regulations; or provide safeguards and security.

6.3.5.2.2 Organization

The Configuration Management Program is a primary responsibility of the Engineering Manager (Design Authority). Groups within Engineering are delegated specific design authority responsibilities. Numerous plant organizations are tasked with implementation responsibilities as illustrated in Figure 6.3-1.

Features (SSCs) controlled by the CM Program include Q and AQ items (see Section 3.8).

AQ items require CM controls as identified in Appendix A of the QAP. CM requirements are reflected in QA and CM procedures.

CM requirements for Q items are addressed in subsequent subsections and paragraphs of this section.

6.3.5.2.3 Key Responsibilities

1. Engineering

   a. Identifies and defines the boundaries for Q systems as a part of the safety analysis documentation and review process;

   b. Performs reviews of plant change reviews (PCRs) for proposed facility modifications and procedure changes, requiring a written safety analysis, unreviewed safety question determination, or decreased effectiveness evaluations;

   c. Provides information derived from the SAR to be used in the development of the Q item database;

   d. Is the Plant Design Authority (DA) responsible for establishing the design requirements, ensuring design output information (documents and data) appropriately and accurately reflects the design input, and for maintenance of the plant's safety basis;

   e. Performs design/modification processes that implement the design control and design change control requirements established in the QAP which includes controls for design inputs, design verification (including analysis software), design changes, design interfaces and design documentation and records;

   f. Manages the CM Program;

   g. Issues the documentation that defines boundaries for SSCs in the CM Program;

   h. Establishes and maintains a controlled database for Q items;

i. Maintains system awareness by reviewing logs, data, work packages and observes operations and performs system walkdowns;

j. Identifies system/component problems and ensures resolution;

k. Evaluates surveillance test results;

l. Monitors PM Program and calibration frequencies;

m. Reviews design proposals and modifications;

n. Ensures that appropriate documents and procedures are updated to be consistent with modifications; and

o. Assists in work package preparation and identification of post-maintenance test requirements.

2. Records Management and Document Control (Plant Services)

a. Develops and operates a Records Management and Document Control (RMDC) program which controls and issues designated documents and acts as the repository with retrieval capabilities for controlled documents and records necessary to maintain the plant's design history; and

b. Maintains an index of documents that are required to be controlled.

3. Procurement and Materials

a. Develops procedures in accordance with the QAP for procurement and control of items;

b. Purchases Q items and replacement parts only from authorized organizations and to the requirements and technical specifications as identified by Engineering;

c. Ensure that only inspected and accepted items are stored and issued for work; and

d. Maintain items in a manner that complies with Engineering issued requirements.

4. Maintenance (Maintenance)

a. Develops and implements procedures to execute a controlled work procedure which provides for verification of data, performance or documentation where specified by the DA; documentation of material used for identification and to ensure design specifications are met; and a process for maintaining equipment history records;

b. Ensures maintenance personnel are knowledgeable of requirements for working on Q systems/items;

c. Perform work only after receiving an approved maintenance work package; and

d. Identifies and furnishes to Documents and Records those work history and traceability requirements for Q items.

5. Operations (Operations)

a. Ensures no modifications are made to design or operational configuration without proper review and approval;

b. Perform and document operational, post-maintenance tests/checks and post-modification tests to assure items are operating as intended;

c. Issue work orders or other authorizations prior to maintenance, testing or modifications activities; and

d. Record the occurrence of tests, calibrations and maintenance activities.

6. Procedures (Training)

Establishes a procedures control program to ensure technical, operations, maintenance and administrative procedures used to apply the CM Program processes are properly developed, received, approved, revised and controlled.

7. Training (Training)

Provides training support to Operations and Maintenance organizations to ensure personnel training is updated to support plant changes.

8. Quality Assurance and Quality Control (Nuclear Safety and Quality)

a. Assists in the development and implementation of the commercial grade dedication process for Q items;

b. Assists in the acceptance process for non-commercial grade Q items;

c. Verifies that DA supplied acceptance criteria is met and that accepted items are appropriately identified;

d. Establishes a program for in-process inspection of maintenance work packages in accordance with acceptance criteria contained in maintenance procedures or provided by the DA;

e. Provides oversight of the problem reporting and resolution program of the Nuclear Regulatory Affairs organization; and

f. Conducts audits and surveillances of processes that implement the CM Program.

6.3.5.2.4 General Modification Process

1. A request for engineering assistance, is reviewed by design engineering management to determine that:

   a. The proposed change is acceptable based upon scope, applicability, justification and/or technical merit.

   b. The change is a substitution and grants approval to continue the substitution process.

   c. The change may proceed through the design process in accordance with engineering procedures and with final approval by the DA and may require additional review by the PORC.

      PORC approval (as described in Section 6.2) is required for changes to the plant as described in the SAR.

2. The engineering design modification process involves:

   a. The assignment of an engineering group to process the engineering request;

   b. The assignment of a design/project team by engineering management. This is comprised of design/project personnel from Engineering (Safety Analysis, Systems Engineering, Nuclear Criticality Safety), Operations, and Maintenance, as appropriate;

   c. A field verification for Q and AQ systems/items that design documentation agrees with as-found configurations. This allows for a technically accurate assessment of the modification's impact on the system;

   d. A 10 CFR 76.68 review performed by qualified personnel (NCS evaluations are described in Section 5.2);

   e. Identification of design documents, procedures, training, post-modification, etc. required to support the modification; and

   f. The approval of appropriate personnel prior to implementation (e.g., DA, PORC).

6.3.5.3 Design Requirements

The SAR, process hazard analyses, and NCSAs are developed based on the plant's known configuration. A reasonable spectrum of postulated accidents that could potentially affect the health and safety of the public and onsite workers are identified and evaluated to determine the risk involved. These evaluations are utilized to identify the SSCs credited for safety and to identify the boundaries of these systems.

Utilizing the boundary definitions and the identification of components responsible for a safety function, the Engineering organization:

1. Conducts a field walk-down, prior to modification, and design document comparison for Q and AQ systems/items to verify the design baseline;

2. Resolves any variances, including the processing of a USQD if appropriate, and generate approved, as-built design documentation; and

3. Utilizes the verified design baseline for future modifications.

6.3.5.4 Document Control

Procedures, documents and records control programs provide for centralized control and issuance of documents critical to the maintenance of the plant configuration and provide a repository for records to verify this maintenance.

6.3.5.4.1 Procedures

The procedure control program assures that procedures are generated, reviewed, approved and distributed in a controlled manner. Reference Section 6.11 for details and clarification.

6.3.5.4.2 Records Management and Document Control

A document control program ensures that approved changes to approved and controlled documents are updated in a timely manner, distributed to controlled copy holders and maintained available to support daily work activities.

Controlled documents, in support of the CM Program, are identified in the procedures that generate the documents and include, but are not limited to, such documents as:

1. Procedures addressing activities affecting SSCs and design features credited with preventing or mitigating accidents as identified in the SAR or other regulatory hazard analyses required by the SAR or QAP.

2. Design documents (e.g., drawings, analyses and calculations).

3. Q item database.

4. Engineering specification data sheets, which include the technical requirements, vendor data requirements and the commercial grade dedication requirements.

5. SAR, TSR, Nuclear Criticality Safety Approvals and other hazard analyses.

6. Procedures and plans addressing emergency operating and response plans.

7. Controlled documents to support maintenance and verification of the design baseline are required by procedures that generate records such as:

   a. Design modification packages;
   b. Acceptance records, for receipt of material, shop and field inspection of work processes supporting maintenance, repair, and testing records;
   c. Maintenance, repair, and modification construction and installation work packages;
   d. Documentation used by Operations to record verification, tests and traceability data;

   Reference Section 6.10 for details and clarification.

6.3.5.5 Change Control

A change control process is documented, which meets the requirements of the QAP and which requires that:

1. Facility modifications, additions, or changes have a 10 CFR 76.68 review performed in accordance with the requirements specified in Section 6.3.2. Information utilized in the 10 CFR 76.68 review includes the following as appropriate:

   a. Systems Requirement Document;

   b. Feasibility studies;

   c. Conceptual design descriptions;

   d. Drawings/specifications;

   e. Other documentation providing a project description and;

   f. Changes in the project.

2. Modifications be evaluated for any required changes or additions to the facility's procedures, personnel training, testing programs, or regulatory documents.

3. Critical repair parts for Q and AQ-NCS items be identified, described, and controlled.

4. Modifications, as appropriate, be evaluated and documented for radiation exposure to minimize worker exposure in keeping with the facility ALARA program, criticality, and worker safety requirements and/or restrictions. Other areas of consideration in evaluating modifications may include: modification costs, similar completed modifications, QA aspects, potential operability or maintainability concerns, constructability concerns, environmental considerations and human factors.

5. Proposed facility modifications receive an independent, technical review which considers the technical feasibility and merit of the proposed change and the identification of appropriate interfaces for inclusion in the change package (e.g., procedures, training, safety).

6. A final review prior to release for operation is conducted which verifies that:

   a. The safety analysis documentation is complete and approved;

   b. Operational procedure changes are completed and other supporting procedure changes have been initiated;

   c. Operational training and qualification changes have been completed;

   d. Design changes are completed and any as-built changes are identified and approved;

   e. Document changes are completed; and

   f. Post-modification testing has been successfully completed.

6.3.5.6 Assessments

The assessment program systematically evaluates the development and effective implementation of the CM Program processes. It assesses the adequacy of the implementation of administrative requirements, the configuration items and their documentation. The assessment program contains the elements of:

1. Field verification of design requirements and documentation. This activity description includes the field verification directed toward the SAR system boundaries and their specific SSCs;

2. Surveillance testing as identified in the TSR and reflected in Operations/Maintenance procedures;

3. Post-modification testing;

4. Administrative procedural audits/surveillances;

5. Activity monitoring; and

6. Problem reporting and corrective action.

6.3.5.7 Training

A training program has been established for workers relied upon to operate, maintain or modify items where required as defined in the SAR. Reference Section 6.6 for details and clarification.

6.3.6 Items Addressed by Compliance Plan

This section is implemented as described with exceptions listed below. The listing of the exceptions also contains a brief description of what is currently in place at the plant. The Compliance Plan provides a description of the exceptions (noncompliances), a justification for continued operation, a description of the actions to be taken to achieve compliance and the schedule for completion of those actions.

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6.3.6.5 Section Deleted

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6.3.7 Flowdown of Certificate Application Commitments

In accordance with 10 CFR 76.35(a)(7), USEC is required to provide 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. One of these managements controls involves the flowdown of certification application commitments into implementing documents.

Implementing documents utilized in the flowdown process shall be controlled in accordance with SAR Section 6.10.2, Document Control Program. Regulatory commitments contained in an implementing document shall be identified and noted as commitments within that document. Changes to an implementing document that contains a commitment are reviewed in accordance with 10 CFR 76.68.

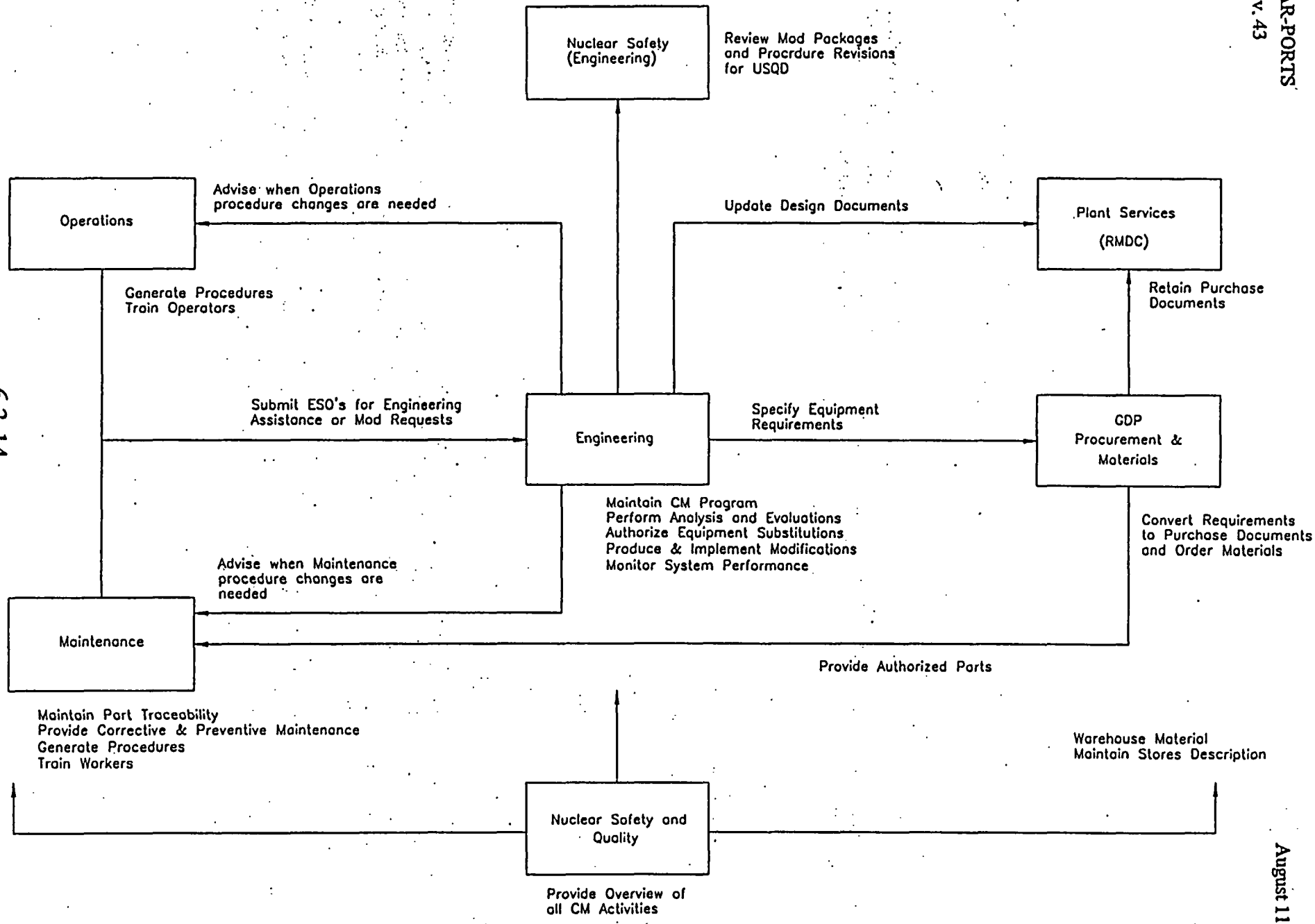

Figure 6.3-1 Configuration Manage[illegible] Functional Relationships

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6.4 MAINTENANCE

The PORTS Maintenance organization provides for the safe, reliable, and cost-effective maintenance of the gaseous diffusion plant. Maintenance for the plant is performed in accordance with Quality Assurance Program and Configuration Management requirements. The maintenance program consists of a mix of corrective maintenance (CM) and preventive maintenance (PM), including instrument calibrations. Trend analysis is used to monitor the effectiveness of the program. Managers have roles and responsibilities that are periodically revised based on corporate goals. Personnel evaluations include the effectiveness of implementation of their responsibilities.

6.4.1 Maintenance Program

The PORTS maintenance program described in this Section 6.4 applies to those Q and AQ NCS structures, systems, and components (SSCs) as identified by the design authority in accordance with Section 6.3, described in Section 3.8 and covered by the Quality Assurance Program. Routine maintenance work is identified, prioritized, planned, scheduled, executed and closed out in accordance with the work control process.

Section 6.4 also applies to AQ SSCs identified by the design authority in accordance with Section 6.3, described in Section 3.8, and covered by the QAP. Identifying, planning, prioritizing, scheduling, and closing out of AQ SSC maintenance is accomplished in accordance with the work control process. For each AQ SSC, the maintenance requirements that must be followed are established based on the impact on the health and safety of the public and workers, and the environment of specific SSCs. The specific graded requirements for the execution of maintenance on each AQ SSC are developed taking into consideration as appropriate (1) the requirements of applicable regulations, codes, and standards; (2) the complexity or uniqueness of an item (or activity) and the environment in which it has to function, as determined by specification, design, or fabrication methods; (3) the history of the item in service, (4) the degree to which functional compliance may be demonstrated or assessed by testing, by inspection, and preventive maintenance methods applied; and (5) the consequence of failure. The configuration management program provides for the identification of AQ SSCs.

6.4.2 Maintenance Organization and Administration

Assuring effective maintenance on GDP identified SSCs requires the following elements:

- Developing policies and procedures for installing, maintaining, and repairing of identified SSCs;
- Performing PM and CM;
- Performing post maintenance testing;
- Performing periodic surveillance activities;
- Identifying required spares, material, and replacement parts in support of maintenance activities;
- Providing technical support for maintenance activities;
- Maintaining the maintenance data system;
- Performing measurement and test equipment maintenance; and
- Implementing on-the-job training (OJT) programs for maintenance personnel.

Implementing these elements is the responsibility of the organizations as described below.

6.4.2.1 Maintenance

The Maintenance Organization conducts corrective and preventive maintenance including calibrations on identified GDP SSCs. Maintenance activities are performed in the following areas; mechanical, electrical, instrument, calibrations, electronics, support shops, and plant services (e.g., maintenance of site vehicles, cylinder handling, janitorial, etc.). Each group manager is accountable to the Maintenance manager for the correct and timely maintenance within his scope.

Maintenance First Line Managers (FLM)/Planners provide the planning function in accordance with the work control process. The FLM/Planners are responsible for the planning of maintenance work activities and for maintaining a record of work accomplished. The work control process includes provisions for:

- Planning tasks to ready-to-work status.
- Ensuring that safety, safeguards, quality, and configuration management are properly and effectively implemented in performance of maintenance.
- Closing Out, including final validation of work performed and review of the acceptance tests performed.
- Providing support and oversight for the Computerized Maintenance Management System (CMMS).
- Providing support for PM scheduling and tracking.

The Maintenance Manager provides oversight of enrichment plant maintenance implementation through self-assessments. The self-assessment program is described in SAR Section 6.8.

6.4.2.2 Operations

Operations provides the integrated scheduling functions for maintenance activities, PM scheduling and tracking support, and TSR/SAR/NCSA surveillance scheduling.

6.4.2.3 Engineering

Engineering provides the technical basis for PM, develops predictive maintenance techniques, evaluates equipment history, tracks and trends performance of designated SSCs, supports root cause analysis of equipment failures, supports reliability studies, provides design criteria, provides acceptance criteria and provides post maintenance testing requirements for identified SSCs. Engineering is also responsible for configuration management and technical support to the Maintenance organization.

6.4.2.4 GDP Procurement and Materials

GDP Procurement and Materials is responsible for procurement, receipt, storage, and issuance of repair parts, materials and components. The organization ensures that all identified SSCs have been inspected by Quality Control, tagged with an inspection tag, and placed in a controlled area. The unit also ensures that identified SSCs are stored in a way that will ensure integrity and availability and that they are maintained according to engineering specifications.

6.4.2.5 Other Support Groups

The Nuclear Criticality Safety, Health Physics/Industrial Hygiene, and Environmental Compliance/Waste Management and Industrial Safety groups provide personnel safety and radiological control requirements needed to perform work safely.

The Quality Control group within Nuclear Safety and Quality performs inspections that are specified in work packages and procurement documents.

6.4.3 Training of Maintenance Personnel

Section 6.6.2 describes the system approach to training methodology used for initial and continuing training.

6.4.4 Maintenance Facilities

Section 3.5 describes PORTS maintenance facilities and specialized equipment.

6.4.5 Procedures

Procedures fall into two broad categories. Administrative procedures define the processes for identifying, scheduling, prioritizing, and coordinating maintenance, as well as prescribing the development of work packages and how maintenance is to be conducted. Technical maintenance procedures or work instructions appropriate for the circumstances are developed for repair, calibration, and testing of specific components or component types.

Section 6.11 describes the procedure control process that ensures procedure development, review, revision, approval, control, and distribution.

6.4.6 Types of Maintenance

Important-to-Safety SSCs are maintained by performing preventive maintenance (PM), corrective maintenance (CM), and TSR/SAR/NCSA surveillances as applicable.

PM are those tasks performed on a periodic basis to prevent failures, facilitate performance and maintain or extend life of the equipment. Predictive maintenance is a type of PM that uses various techniques to track and trend equipment performance in order to predict failures. PMs are identified, scheduled and performed in accordance with maintenance procedures, engineering procedures or work instructions, as appropriate. Reports are available showing the status of scheduled PMs. PMs are adjusted as described in Section 6.4.7.

CM are those actions to check, troubleshoot, and repair equipment that has degraded or failed. CM identification, prioritization, planning and scheduling, performance and documentation are discussed in Section 6.4.7.

TSR surveillances are performance checks, calibrations, tests, and/or inspections which are performed to verify the proper operability of equipment and/or systems in accordance with the plant's Technical Safety Requirements. The Technical Safety Requirements identify the surveillance requirements and frequencies.

6.4.7 Maintenance Programs

The Maintenance Program is comprised of CM, PM, TSR/SAR/NCSA surveillances and trend analysis. Work control is the process used to implement elements of the maintenance program. Key elements that are used to help the plant achieve the desired level of program performance are listed below.

CM consists of the following elements:

- Work Request - A Work Request is the mechanism by which the need for maintenance is identified and prioritized. After review, work requests are changed to work orders to plan, schedule and perform work.

- Planning and Scheduling - Planning and scheduling activities result in: (1) A work package which identifies the scope, procedures/work instructions, necessary permits and required materials; (2) identification of required support services; and (3) identification of the need for coordination with other jobs. Maintenance activities are prioritized by the facility operational manager, or designee, based on safety significance and available resources.

- Work Execution - Work packages are reviewed, a pre-job briefing is conducted in accordance with procedures, work is conducted as planned, and the maintenance that was performed is documented.

- Post Maintenance Testing - Provides for testing to ensure that equipment and/or components will fulfill their required function when returned to service after maintenance.

- Work Order Completion - Provides a post maintenance package review to validate job package completion.

PM consists of the following elements:

- Basis for PM - The bases for preventive maintenance tasks are developed through a review and applicability evaluation of manufacturer recommendations, and available industry standards, and with historical operating information. PMs are coordinated by Engineering and Maintenance and involve input from design engineering, systems engineering and operating/maintenance personnel as appropriate. The formal documented bases for the tasks are developed, evaluated and approved by Engineering.

- Scheduling - PMs are scheduled according to plant conditions, and task frequencies. The work control process identifies SSCs that require PM work packages.

  PMs are conducted and closed out in accordance with the work control process and PM program procedures. The TSR surveillances that are also credited in the PM program will be conducted as required, or the actions required by the TSR will be followed. For PM's beyond the grace period (not including TSR surveillances), Engineering with concurrence from the applicable functional area manager, or designee, must determine the effect of extending the PM beyond the maximum interval (as stated in Section 3.0 of Definitions) on system operation. The decision includes a determination of compensatory actions and an acceptable completion date for the PM tasks.

- Adjustments - Changes to, adding or deleting of PM tasks are performed according to the appropriate procedures and may be recommended by operations, maintenance or engineering. Changes to tasks may be warranted by a system's performance compared to the performance indicators established for that system. Changes may also be initiated if the level of corrective maintenance dictates that additional preventive measures are necessary to prevent premature failure of a system or its components or as found conditions indicate that PM is occurring more often than necessary.

TSR surveillances are performance checks, calibrations, tests, and or inspections which are performed to verify the proper operability of equipment and/or systems in accordance with the plant's TSRs. The TSRs identify the surveillance requirements and frequencies.

Trend analysis is the collecting and documenting of equipment performance history and maintenance data that is used to trend the reliability of SSCs. Equipment history is collected in the work package. The data is used in the evaluation of PM task and frequencies, for tracking and trending recurring problems, as a basis for determining material condition, and evaluating equipment performance and reliability.

6.4.8 Section Deleted

6.4.7.1 Work Control Process

The Maintenance organization owns the CMMS and Primavera software and provides database administration for these systems.

Maintenance Planning starts with the initiation of a Work Request (WR), including priority, identifying the need for maintenance. The Work Request is reviewed by Maintenance for determination if a work package is required. The Work Control process identifies SSCs that require work packages before work can be started. The work control process procedure also contains allowances for non-safety SSCs requiring work packages.

The composition of a work package is dependent on the maintenance to be performed and is defined in the work control process procedure. The minimum requirements of a work package for safety system maintenance includes: task description; work instructions or procedures; and post-maintenance test requirements. Depending on the scope of the work, the package may also include equipment specific procedures or checklists and quality control inspection, nuclear criticality safety, radiation protection, OSHA, or operational requirements.

The FLM/Planner completes the work package using the approved work control procedure that lists the criteria for support group reviews by Engineering, Security, Fire Services, Health Physics/Industrial Hygiene, Nuclear Criticality Safety, Quality Control and Environmental Compliance/Waste Management and Industrial Safety prior to starting work. Any temporary or permanent modification to SSCs requires Engineering review and approval according to Engineering procedures. The FLM/Planner also coordinates support group involvement in accordance with procedures.

The applicable Maintenance manager reviews work packages for technical content, accuracy and completeness, and provides written approval to use the package prior to starting work. Conflicts in priorities between operating organizations are resolved through the work control process.

Permanent or temporary modifications and replacements are performed in accordance with the requirements described in Section 6.3. These procedures describe the mechanisms for the initiation and control of replacements, additions, or modifications (temporary or permanent) to plant systems or equipment. They delineate the individual and group responsibilities, and prescribe the records and reports that shall be produced and/or maintained at the plant for documenting these changes.

Authorization for release of equipment or systems for maintenance is granted by designated operations managers. Such authorization is documented in the work package and is based upon verification that the equipment or system may be taken out of service. The work package identifies other needed permits. The affected equipment or system is isolated to provide protection for plant personnel

and equipment. Radiation Work Permits, lockout/tagout, and various safety and health permits establish conditions to ensure personnel and equipment protection.

When entry into a closed system is necessary, the foreign material exclusion (FME) control process is utilized in accordance with procedures to prevent entry of extraneous material and to ensure that foreign material is removed before the system is closed.

SSCs that are replaced during maintenance are compared to the installed part and verified to be approved for installation. Temporary modifications, such as temporary bypass lines, electrical jumpers, lifted electrical leads, and temporary trip settings, are governed by approved administrative procedures.

When equipment is returned to service, Operations will verify and document the satisfactory completion of post-maintenance testing and place the equipment in operation.

6.4.9 Post Maintenance Testing

PMT requirements are approved by and may be developed by Engineering. As needed, special PMT requirements are requested on a case-by-case basis. Performance of PMT is documented in the work package.

6.4.10 Procurement, Receipt Inspection, Control, and Issuance of Repair Parts, Materials and Services

The Quality Assurance Program describes the requirements for procurement control, and the control of purchased material, equipment, and services. Procurement of items is performed in accordance with corporation and plant procedures.

Repair parts, components and material requirements for Important-to-Safety SSCs are listed on the engineering approved specifications. The engineering approved specifications and associated inspection plans provide the design criteria and inspection requirements needed when procuring SSCs.

The buyer obtains the latest engineering approved specifications and inspection requirements and reviews them for changes. Commercial grade items are procured according to catalog specifications from the manufacturer or factory authorized dealer or distributor. A computer-based procurement system identifies materials which require inspection upon arrival at the plant. Upon receipt, the item is placed in a segregated area for inspection and acceptance. Quality Control is provided an inspection package by the Buyer to determine acceptability of the item. A unique identification number is placed on the item for traceability. If the item is rejected by Quality Control, it is placed in a segregated area or tagged until the nonconformance is dispositioned by the appropriate Responsible Disposition Authority.

Traceability of Important-to-Safety SSCs is maintained when they are received and placed in stores for use. Configuration management provides for parts traceability after they are installed in the plant.

6.4.11 Control of Measuring and Test Equipment

To maintain accuracy within specified limits, the maintenance program requires that M&TE be properly controlled, calibrated, and adjusted at specified periods in accordance with program procedures.

The following items are included in PORTS procedures:

- A unique identifier
- Calibration intervals defined and entered into a recall system
- A label to indicate calibration status
- An inventory listing of controlled M&TE
- Evaluation of calibrations using M&TE that is subsequently found out of tolerance
- Preparation and maintenance of calibration records
- Measures for the storage and control of M&TE.

M&TE is calibrated in accordance with procedures. Standards used to calibrate devices have the required accuracy, stability, and range for the intended use and are certified and traceable to the National Institute of Standards and Technology. If no national standards exist, the basis for calibration is documented. The M&TE program will apply to all M&TE at PORTS.

6.4.12 Maintenance History

CMMS and completed work packages are used to capture maintenance history for Important-to-Safety SSCs. The maintenance history includes the items discussed below.

6.4.12.1 Component Identification and Description

Components are identified by name, number and component identification. Where available, the description includes the manufacturer's name, model, serial number, and quality classification. Additional reference may be made to purchase order number, vendor manuals, drawings, system logic and/or flow diagrams, and applicable maintenance procedures. Engineering procedures specify control of modification documents.

6.4.12.2 Maintenance Record

The Equipment History Form is a record of significant work performed on the component and is included as part of the work package as appropriate.

6.4.13 Section Deleted

6.5 OPERATIONS

The site is large enough to provide a considerable buffer between the enrichment process and our rural neighbors. Plant operations are continuous with coordination of operations performed by the Plant Shift Superintendent (PSS) from a central control facility. The plant is protected on a continuous basis by fire services and Protective Forces. Each significant building is equipped with fire alarms and water sprinklers. Emergency mutual assistance exercises are conducted biennially with the emergency forces of the state and of the surrounding communities.

There is a public warning system to alert neighbors in the event of any plant problem that might affect them. All major liquid effluent points are guarded by impoundment ponds to intercept and contain spills. There also are internal impoundment structures, spill control equipment, and monitoring stations with alarms. The principal toxic gases on the site are the uranium hexafluoride process gas, fluorine, chlorine trifluoride, chlorine and hydrogen fluoride. Liquid hazardous chemicals include oil, nitric acid, sulfuric acid, and a variety of other chemicals in smaller amounts.

The plant is normally in one of three modes of operation, from a safety perspective.

Normal Operations

Most of the time is spent in normal operations; in this mode the following conditions apply:

- Operations are proceeding within expected parameters with no safety impacts from deviations,
- Technical Safety Requirements (TSRs) are in effect,
- Routine effluents or emissions within permits and certificate conditions with no significant impact to the public or environment, and
- Personnel exposures are below 10 CFR 20 limits and OSHA requirements.

Off-normal (but not emergency) Operations

Occasionally, process upsets and/or equipment failures occur which result in "off-normal" conditions within localized areas of the plant; these "off-normal" modes are as follows:

- Small releases of UF_6 or other toxic gases (such as F_2, HF, CLF_3, Cl_2) that result in evacuation of the immediate area and monitoring for reentry, but do not affect other areas of operations of the plant and have no impact off site.
- Occupational safety injuries and/or illnesses with a response required to render aid or transport to the plant or off site medical facility,
- Small fires that are quickly extinguished,

- Unexpected radiological contamination that requires reporting of plant areas or additional employee protective measures.

These "off normal" conditions are managed by the PSS from the X-300 Plant Control Facility with involvement by plant shift emergency response and/or Health Physics personnel. It may involve call-in of Industrial Hygiene or Safety personnel.

Emergency Operations

The third mode is an emergency, as described in the Emergency Plan, which involves an "Alert" or "Site Area Emergency" declaration and activation of the Emergency Operations Center.

The remainder of this section provides an overview of the major operating areas of PORTS with a brief discussion of the safety and safeguards risks and the controls and operational surveillances in place to manage these risks. A description of the plant and plant operations is provided in detail in Chapter 3; a detailed accident analysis and discussion of risks associated with plant operations is provided in Chapter 4.

The X-300 Plant Control Facility

The process is controlled primarily in the three Process Buildings. Each individual cell of equipment in those buildings has a Local Control Center (LCC) where fine control of the equipment is exercised as needed. Alarm functions and higher level controls also are located in Area Control Rooms (ACRs) within each process building. The most important alarms, process data and control functions also are routine to the X-300 Plant Control Facility (PCF). (Therefore, emergency process functions may be performed at the cell control centers (LCCs), the ACRs or the PCF.) The personnel within this central facility provide the coordination of cascade operations and other key functions and the initial response to all plant emergencies. The building also contains the readouts to the plant's nuclear criticality alarm system (neutron detectors).

The Feed Facilities

Normal assay uranium hexafluoride feed and Paducah Product feed (nominal 1.9 percent U^{235} assay) are fed to the enrichment cascade from the X-343 Building (feed capability also exists in the X-342 Building). The UF_6 is heated with steam in containment autoclaves, converting the solid UF_6 in the feed cylinders to a liquid-vapor equilibrium. The material then flows as a gas into the process equipment in the X-330 and X-333 Process Buildings through steam-traced, insulated headers.

The X-343 facility is equipped with an evacuation system for the purpose of evacuating small quantities of UF_6, purge gases, or light gases from the cylinders in autoclaves and associated piping. This evacuation is accomplished by valving the system to be evacuated into a feed header, the X-333 evacuation line, or the X-343 cold trap/chemical trap system. The feed or evacuation line returns the gases to the cascade. The cold trap/chemical trap system traps residual UF_6 and vents lights to atmosphere.

The breaching of a liquid UF_6 cylinder is one of the plant's worst accident scenarios. Consequently, the autoclaves are provided with a variety of sensors and containment devices to prevent the release of UF_6 while heating a cylinder, and the buildings are equipped with nuclear criticality alarms. Cylinder lifting devices and practices are also designed too prevent dropping and rupturing a liquid UF_6 cylinder. The UF_6 feed operations are conducted under the control of Technical Safety Requirements.

The Process Buildings

The process equipment is housed in three process buildings: X-333, X-330, and X-326. The largest equipment is housed in X-333, where the majority of the power is consumed and the majority of the UF_6 is processed. Two smaller equipment sizes are operated in the X-330 building, which contains both higher and lower assays than the X-333 Building. The smallest process equipment is located in the X-326 Building, where light gases-primarily nitrogen and oxygen, although other low molecular weight gases may be present at any given time-are vented from the top of the cascade through small "purge cascades" to the atmosphere. Chemical traps are utilized as a part of the purge cascade to remove residual uranium which may be contained in the vent stream. Many of the higher assay product withdrawals are made at the bottom of the X-326 Building from 27-size equipment. However, most of the equipment in the X-326 Building is in permanent shutdown. Only a relatively small amount of equipment at either end of the building is still operated -- the purge at the top and a portion of the X-27 equipment at the bottom of the building. The potential hazards associated with the process buildings are directly related to the UF_6 inventory in the enrichment operations and in associated support activities. These hazards may be summarized as UF_6 releases, nuclear criticality accidents, exothermic chemical reactions, fires, and process emissions. Enrichment process operations are conducted under the safety controls identified in the Technical Safety Requirements.

UF_6 Releases

The pressures in the process equipment vary with power level. Except at the very low plant power levels (less than about 1300 MW), much of the X-333 equipment and the 31-size cells in the X-330 Building are operated at above atmospheric pressures (typically 15-20 psia). Such cells are equipped with release detection sensors monitored by a computer system. These stages also have a special seal design to reduce the likelihood of a release to the atmosphere.

Nuclear Criticality Safety

Nuclear criticality alarms are located throughout the process buildings. Nuclear safety is enhanced by special equipment features. The process equipment is sized such that a gas phase criticality is not possible. However, solid masses can result from the following conditions. Inleakages of ambient air may form solid UO_2F_2 by the reaction of the process gas with the moisture in the air, and the UF_6 can freeze out following certain equipment failures. Both of these conditions usually are readily detected because they cause operational problems. Additionally, gamma monitoring is performed on a routine basis to detect deposits. Solid deposits in the enrichment cascade cannot, of themselves, result in a criticality accident; moderation and unfavorable geometry are also required.

Moderation in the cascade is controlled in part by the process chemistry. Any inleakage of moist air produced the UO_2F_2 hydrolysis product and hydrogen fluoride. Although some of the hydrogen fluoride may be adsorbed on the UO_2F_2, most of it is swept away immediately by the flow of process gas and ultimately vented through the purge cascades. Entrance of bulk water to the cascade is prevented by the design of the cooling system, which isolates the process gas from the cooling water by intermediate heat exchange with Freon-114 coolant. The Freon is evaporated in the UF_6 cooler located in each stage. The Freon vapor then flows to a condenser where it is condensed by cooling from the plant recirculating cooling water system; the liquid Freon then flows by gravity back to the stage cooler to repeat the cycle.

Exothermic Chemical Reactions

There is a potential exposition hazard in the purge cascades. If procedures are not followed Freon from cooler leaks can accumulate in the purge equipment and react with the oxidant gases used for cell treatments. A Freon Degrader is used to reduce Freon concentrations in the purge cascades. Also, the concentrations of both chemical species are monitored carefully and administrative controls are used to prevent the occurrence of unsafe concentrations. Exothermic reactions also can occur in the lower cascade, and are prevented by engineered and administrative controls.

Fires

The process buildings have large quantities of lubricating oil that presents a significant fire hazard potential due to the combustible loading even though the oil has a high flash point. All the process buildings are protected from fire by water sprinkler systems.

Process Emissions

Light gases also are vented from the X-330 and X-333 Cold Recovery Facilities and from both the X-333 and the X-330 Wet Air Evacuation Stations. The Cold Recovery Facility recovers uranium from gases by cold trapping, and the lighter gases may then be vented to the atmosphere through chemical traps. The Wet Air Evacuation Stations vent gases that contain very low concentrations of uranium through chemical traps to the atmosphere. Vent gases from Cold Recovery, Wet Air, and the purge cascades are all monitored continuously for uranium by ionization chambers called "space recorders." Composite monitoring also is performed continuously at these vents by a isokinetic sampling system utilizing special chemical traps. The composite trap samples are analyzed on a weekly basis for uranium, technetium, and fluorides.

Product and Trails Withdrawal Operations

The withdrawal of UF_6 enriched product is accomplished in the X-333 Low Assay Withdrawal (LAW) facility and the X-326 Extended Range Product (ERP) facility. Withdrawal of depleted uranium is accomplished at the X-330 Tails Withdrawal facility. The tails and product withdrawal facilities compress the UF_6 gas, which is then cooled to a liquid and drained into cylinders. The cylinders are moved from the withdrawal position to cool-down locations where the cylinders remain until the contents are solidified. These operations are conducted under the safety controls specified in the Technical Safety Requirements.

Cylinder Sampling and Transfer Operations

Enriched 10-ton product cylinders (parent cylinders), cooled to solidify the UF_6 are moved to the X-344 Building for sampling and transfer to 2.5-ton customer cylinders (daughter cylinders). These operations are performed in containment autoclaves, just as in the X-343 Building. In addition, incoming product and feed cylinders are sampled in the X-343 Building in autoclaves.

The X-344A Facility is equipped with an evacuation system for the purpose of evacuating small quantities of UF_6, purge gases, or light gases from the cylinders in autoclaves and associated piping. This evacuation is accomplished by valving the system to be evacuated into the X-342 PG tie-line or the X-344 cold trap/chemical trap system. The tie-line returns the gases to the cascade. The cold trap/chemical trap system traps residual UF_6 and vents lights to the atmosphere.

The breaching of a liquid UF_6 cylinder is one of the plant's worst accident scenarios. Consequently, the autoclaves are provided with a variety of sensors and containment devices to prevent

the release of UF_6 while heating cylinder, and the buildings are equipped with nuclear criticality alarms. These operations are conducted under the safety controls specified in the Technical Safety Requirements.

The X-342 Building contains the fluorine generation system for the plant. Liquid hydrogen fluoride is converted to hydrogen and fluorine in electrolytic cells. The building is equipped with release alarms. The consequences of a potential fluorine or HF release are minimized by using small (nominal 850 pounds HF) HF cylinders to feed the electrolytic cells and by storing the minimum amount of fluorine necessary for a reliable process supply -- usually 1000 to 2000 cubic feet at a nominal 45 psia pressure.

The X-705 Decontamination and Uranium Recovery Facility

The X-705 Building contains a large decontamination spray "tunnel" and manual cleaning facilities for the decontamination of failed process equipment. Citric acid and nitric acid are the primary decontamination chemicals. A solvent extraction process and calciners are used to recover uranium from high concentration solutions and convert it to oxide.

The raffinates from the solvent extraction columns are treated with caustic to raise the pH and precipitate the uranium and heavy metals. The solutions may then, if necessary, flow through ion exchange columns to remove technetium-99. Finally, the solutions are treated in a biodenitrification process to remove nitrates, and the treated waste flows to the X-6619 Sewage Plant.

A microfiltration system is used to remove uranium from dilute wastes. The dilute solutions are pH adjusted to precipitate the uranium, and the very light sludge is processed through microfiltration columns to remove the uranium solids. Redundant pH sensors for the solutions and pressure drop sensors for the filters provide nuclear safety by preventing uranium from reaching nonfavorable geometries.

The primary facility safety issue in X-705 is nuclear criticality. Most of the equipment and solution storage columns in the building are of geometrically favorable design to provide nuclear safety. Redundant chemical analyses are used to provide safe batching of wastes and solutions in nonfavorable geometries. Building drains are sealed to prevent spilled solutions from reaching nonfavorable geometries, and to prevent spilled treatment chemicals and radionuclides from reaching the environment and the public. Operations in this facility are conducted under safety controls as specified in the TSRs and NCS program documentation.

Other Buildings

There are several other buildings that deserve mention. The X-700 Building contains cleaning chemicals and radiation sources used for instrument calibration. The building drains are isolated to provide spill control. The radiation sources are contained in engineered structures with alarms to guard against personnel exposure.

The X-710 Laboratory contains a variety of laboratory chemicals and small amounts of process material. It is equipped with conventional laboratory hoods and conventional industrial spill control measures are maintained.

All buildings that contain fissile material are provided with radiation clusters and warning horns.

The remainder of this section describes the operations associated with the above facilities and activities.

6.5.1 Shift Operations

The gaseous diffusion process (GDP) is designed for continuous operation; the function of the GDP is complex and requires significant planning for interruptions and shutdown of equipment. To support this need for continuous operation, a work force is required 24 hours per day.

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 day shift provides the administrative support, activities such as design and fabrication where around the clock effort is not required, procedure 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 continued plant operation and the evolutions, exchange of information, and response to abnormal and unusual conditions necessary to ensure safe and efficient plant operation. Typical activities of the shift include: provide oversight and direction for all plant operations; monitor systems and equipment for proper performance; conduct routine back shift maintenance and emergency equipment repair; prepare equipment for day shift repair/preventive maintenance functions; and respond to emergency situations.

Operational activities of the plant are controlled by the on duty PSS whose normal watch station is in the X-300 Plant Control Facility (PCF). The PSS, or designee, functions as the Plant Emergency Director/Incident Commander for emergencies until relieved and reports directly to the Crisis Manager. Emergency command and control is described in the Emergency Plan.

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 offsite communications capabilities are located in the PCF. The plant power system is monitored and controlled through a communication network with the power supplier. Typical operational activities that are monitored and controlled from the PCF include determining and establishing optimal plant power level, executing or altering the maintenance work plan if necessary, and maintaining the necessary manpower level to support plant operations.

Staffing levels for the shifts are not fixed but are based on the expected or planned activity for the shift period. Staffing levels take into account the routine monitoring of plant equipment including operator rounds, expected operational activity level, facility size, and Technical Safety Requirements (TSR) specified staffing requirements. When special activities are included in the work plans, the staffing will be increased as required to perform the planned activity. The average shift staffing on backshifts is approximately 116. The TSR required minimum staffing level is listed in Section 3 of the TSRs. This is a small fraction of the average shift staffing.

Each shift organization is composed of a PSS and an assistant PSS; a cascade controller (CC) who directs overall cascade activities; first-line managers for the cascade buildings, power operations, chemical operations, and utility operations, health physics technicians, Security Shift Commander, Fire Services Shift Commander, operators, security patrol officers, and firefighters. Less than this normal shift staffing is permitted for short periods with the concurrence of the PSS to allow for call-ins or other compensatory actions.

The PSS provides a direct chain of command from the Shift Operations Manager, Operations Manager, Transfer and Shipping Plant Manager and General Manager to the shift operating staff, and serves as the senior shift manager in directing activities and personnel. The operations line organization is accountable to the PSS for reporting plant status.

The CC provides managerial oversight, operations coordination, and assures adequate staffing for all cascade operations on a 24-hour basis. This person approves, directs, and integrates all significant cascade operational activities under the oversight of the PSS.

The remaining members of the shift organization perform the needed functions for round-the-clock operations. The assistant PSS supports the PSS in management during shift operations. The first-line managers provide management for, coordination of, and assurance of proper execution of assigned tasks. The Shift Engineer provides engineering support for technical issues involving Operations. Health physics technicians provide support for 24-hour shift operations. The Security Shift Commander supervises the activities necessary to ensure the protection of plant facilities, government property, and classified information. The Fire Shift Commander supervises shift fire services work activities and responds to plant emergency events.

There are many diverse systems for operational communications. Commercial telephones, an internal plant telephone system, radio networks, a plant public address (PA) system, emergency signals, and a pager system are available to provide necessary communications in operating the plant. The PCF is the focal point for all emergency reporting and initiating of all emergency responses. A special emergency telephone network is available in the PCF. Fire alarm and sprinkler indicator systems, and criticality alarm panel, as well as numerous operational alarms are monitored. As described in the Emergency Plan, the PSS will initiate offsite notifications and plant personnel call-ins when required.

In accordance with the corrective action program, plant personnel are required to report abnormal events or conditions that may have the potential to harm the safety, health, or security of on-site personnel, the general public, or the environment. Plant personnel are also required to immediately report conditions which may require emergency response. The PSS reviews potentially reportable or inoperable safety system equipment reports and determines proper disposition.

6.5.2 Operations

The cascade is the UF_6 enrichment portion of the plant. The cascade is composed of three major process buildings which house two parallel enrichment cascades that share common product and tails withdrawal facilities. There are auxiliary facilities such as the recirculating water pump houses which are also under the direct control of cascade operations.

The Operations Manager is responsible for overall operations. This includes operation of cascade equipment, planning for power usage, control of feeds, product and tails material including sampling, operating plant utilities, radiological decontamination, equipment cleaning, uranium recovery, and operation of plant laundry. The Operations Manager is supported by managers in the following groups:

Shift Operations, Chemical, Utilities, X-340s, and Cold Standby. These group managers have subordinate managers assigned to functional areas to provide oversight of the day shift operations. The Work Control group and Power Operations Section also support the Operations Manager.

The optimum cascade arrangement for specific power levels, product and tails assay levels, and feed availabilities is determined by the day shift operations staff. The rotating shift organizations follow daily instructions and work plans developed and communicated by the group managers. The cascade controller/coordinator has the responsibility to change cascade related priorities should the need arise. Changes to plant priorities for activities on shift require the approval of the PSS.

The Work Control group is responsible for scheduling of maintenance work activities and providing overall integrated scheduling of projects. The scheduling function of the work control process includes provisions for:

- Coordinating work identified by requestor.
- Scheduling work to be performed.
- Executing, including coordinating work force and support groups and ensuring communications flow between work groups and management.
- Scheduling and tracking of TSR/SAR/NCSA surveillances.
- Scheduling and tracking of preventive maintenance.

The Cold Standby Manager is responsible for overall operation and maintenance of the cascade in the cold standby mode, including deposit remediation activities, and for ensuring the available staff is adequately trained to perform all related tasks.

The Utilities Manager, in conjunction with key building managers, provides the plant with sanitary water, chilled water, steam, air, nitrogen, and sewer services. These must be supplied on a continuous basis to meet the cascade requirements. Any outage is coordinated with customers to assure proper planning to provide temporary services as necessary.

The Power Operations Section Manager is responsible for managing the power supply system as well as being involved in the activities associated with power contract and power scheduling. Power systems rely on scheduled preventive maintenance activities to ensure a dependable supply. The power facilities are operated and monitored 24-hours per day and, therefore, have a shift organization which interfaces directly with the PSS.

The Chemical Operations Manager is responsible for operation of the plant laundry, nonradiological and radiological decontamination, cleaning of respiratory protection equipment, and uranium recovery.

The X-340s Manager is responsible for UF_6 cylinder handling within the plant including product cylinder shipment, and operating the plant fluorine facility.

Other organizations within the plant provide support such as supply of spare parts and equipment, handling of scrap and waste, provisions for employee safety, necessary analysis to control operation and

protect the environment, provide status of equipment design, systems engineering support and design change, and administrative support. Chapter 3 provides additional detail on specific cascade equipment and support systems.

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6.5.5 Operator Responsibility, Authority and Shift Routines

Plant operations, shift routines, and operator responsibilities are activities that are governed by procedures at PORTS.

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Key elements are summarized as follows:

- Alarm Failures and Deactivations — Guidance provides the administrative requirements for taking an alarm out of service including, if necessary, evaluating operability and control of restrictions imposed if continued equipment operation requires compensatory actions.
- Shift Routines — Guidance is provided for on-shift operators to perform inspection tours, shift turnover, dealing with abnormal operating parameters, etc.
- Operating Area Logs and Records — Guidance is provided for filling out required logs and other required records.
- On-Shift Training - The On-Shift Training program is conducted in accordance with applicable plant training procedures and includes required reading and group briefings. The program provides a documentation record of whether or not an individual has completed and assignment required for performance of their job.
- Daily Operating Instructions and Long Term Orders - Guidance is provided for documents that contain instructions issued by management.
- Control of Operator Aids — Guidance provides administrative controls to ensure only appropriate information is used, operator aids are kept up to date and removed when no longer needed or correct.
- Deficient Material Condition Tags — Guidance provides a means of identifying to other personnel when a maintenance work request has been issued to correct a plant deficiency such as a leak or a broken component.
- Control Room Activities — Guidance provides control over the activities performed in the "at-the-controls" areas.
- Organization and Administration — Guidance defines the organization of the Operations Organization and defines the duties and responsibilities of each position.

6.5.5.1 Operator Responsibilities

Authority to operate equipment under the responsibility of the Operations Organization requires qualification. Plant qualifications for operational positions that could affect nuclear safety require training and successfully passing required testing as described in the appropriate administrative guide for the required position. The training program is described in Section 6.6 of the SAR. This proficiency is routinely enhanced by retraining and testing coupled with on-the-job training. Training is based on operating procedures and their use is a necessary extension of training. Some areas require special qualifications before job performance is allowed. Examples of positions in cascade operations with special qualifications include control room operators, cold recovery operators, and operators in UF_6

handling facilities such as X-343, X-333, and X-330 and X-326.

In Chemical Operations, qualification is required for large/small parts disassembly/decontamination, decontamination and recovery, field decontamination and cleanup, wastewater treatment, and cylinder washing operations.

Operator rounds are a key part of overall operator duties. These activities have been proceduralized to identify requirements. A listing of items to be checked on the rounds is included in the applicable procedures.

Training and procedures define the appropriate response by operators to alarms. In areas important to nuclear safety, a set of alarm response procedures has been developed which identifies the alarm signal and the proper response.

6.5.5.2 Shift Routine

Access to the at-the-controls areas in the area control rooms (ACR) is limited to avoid confusion, prevent blocking visual access to operating panel indications, and enhance control room operator functionality. The at-the-controls area applies to ACRs in X-326, X-330, X-333, and the control area of the PCF.

The training program for an operator includes an on-the-job training (OJT) program. On-shift training is conducted in accordance with applicable plant training procedures. Only qualified personnel may perform operational evolutions unsupervised. Whenever trainees operate equipment, a first-line manager or qualified employee observes and assists the trainee to ensure errors are not made that could adversely affect safety.

An essential part of the shift routine is operator rounds. In facilities subject to continuous operation, 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. For many key systems, acceptable operating parameters are listed on the round sheet to aid the operator in ensuring system operation remains within the design basis. "Batch work" such as removing/restoring a cell to service in the cascade require additional rounds to be conducted prior to the operation and in accordance with applicable procedures.

Changes to operations and procedures are disseminated to all personnel through the required reading program and/or through required training. Operators are required to review revised procedures and instructions that affect their duties prior to resuming watch standing duties. Management provides day-to-day guidance on plant operations and temporary changes to operational instructions via a controlled system of shift operating instructions.

Formal communication of operating information is defined in plant procedures. Preparation for turnover, the actual turnover, and turnover follow-up are included in these procedures. A sign-off is required to signify understanding by all parties involved.

6.5.5.3 Logkeeping

Operator logs provide a means of documenting the information on activities occurring during shift. Logbooks are maintained in the ACRs, shift/first-line managers' offices, CC's desk, and in the PSS office and in other designated areas.

Logbooks maintained in ACRs record chronological operations of the shift and equipment realignments and problems encountered. Supervisors' logs are often the summation of activities from different operating areas under their supervision. Therefore, the chronology of events is not as rigorously followed except under emergency conditions. Entries in all logs are in black ink with corrections lined out, initialed, and dated. Logs are essential in completing shift turnover, and turnover is documented by signatures in logs by the outgoing and incoming personnel. This formal turnover is done to assure the information necessary for smooth activity transition is exchanged. Log books are also routinely reviewed by day shift personnel, individual building managers (called facility coordinators), group heads, and others. These reviews are indicated by the reviewer's initials in the logbook.

6.5.6 Operations Procedures and Operator Aids and System Labeling

Operating procedures are prepared, reviewed, and approved as outlined in Section 6.11. Shift operations personnel follow and use procedures as described in Section 6.11. Actions required by emergency and alarm response procedures may be completed before referring to the procedure if conditions warrant.

Another valuable tool for support of operator duties are the operator aids. These aids consist of posted sketches, notes, graphs, instructions, and drawings. A formal program is in place to control and account for these aids and ensure they are correct in all respects. Operating information is also disseminated using organization memos, daily instructions, and long-term shift orders. These communications provide information not included in procedures which is required for a special short-term period by the operations personnel.

Major equipment and piping systems are labeled. These labels aid personnel in identifying specific equipment and systems in the field. In the large cascade buildings, the grouping of equipment into units and cells is clearly marked so that the possibility for operator error is minimized. In addition, equipment location can be identified by a column grid system (alpha-numeric) which can be used to direct operators or emergency responders to the proper location.

6.5.7 Permits and Tagging

A lockout/tagout permit system is in place which provides controls for ensuring safe isolation of systems for maintenance or testing. Lockout/tagout activities are performed by authorized personnel and isolation points are independently verified. A "Caution Tagging" system has been implemented as a warning to operators to stop and seek assurance of proper action before proceeding. Section 6.4 discusses this in more detail.

6.5.8 Management Monitoring of Operations

A management assessment program is used by management to provide oversight of plant operations. The program includes managers at all levels and is described in procedures. The results of the program are disseminated to appropriate management for evaluation.

6.5.9 Control of Equipment

The Cascade controller/coordinator provides overall coordination of operational activities including removing/returning of cascade equipment to service, increasing/decreasing power levels, radiological protection, maintenance, and instrumentation to accomplish the operating objectives for the shift and to maintain safe and efficient operations. Communications between the controller/coordinator and the first-line managers who authorize the removal and return of equipment and systems from service for maintenance, testing, or operational activities is conducted using the communications methods described above.

Qualified operators are capable of diagnosing facility and equipment conditions and are authorized to perform tasks during normal, off-normal, and emergency conditions which may include the shutdown of equipment. Restart of equipment from routine shutdowns is authorized by the first-line manager. The PSS is authorized to stop operations when system operability or the overall safety of operations is in question. The PSS is also authorized to initiate restart after shut down for non-routine reasons. For shutdowns that are directed by the Vice President, Operations; Nuclear Safety and Quality Manager; General Manager or the Transfer and Shipping Plant Manager; the PSS may authorize restart only after obtaining the approval of the Transfer and Shipping Plant Manager (who will in turn obtain the necessary concurrence as described in Section 6.1.1.9).

For the purposes of ensuring the ongoing operability of TSR- required equipment, Operations utilizes procedural requirements which include the following activities:

- Day-to-day operation of the facility
- Plant walkdowns or tours
- Operator observations
- Inspections, assessments, and audits
- Engineering evaluations and design reviews
- TSR-required surveillance tests

The above activities combine to provide continuing assurance that TSR-required equipment will perform as required, when needed.

To regain operability of equipment which has failed, been taken out of service, or had maintenance performed on it, each of the following elements must be satisfied:

- Applicable surveillance test(s) must be successfully performed
- Any involved technical support organizations, such as engineering or criticality safety, must approve restart or return to service
- PSS gives authorization

When all of the above elements have been satisfied, the first-line manager can return the equipment to service.

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6.6 TRAINING

In accordance with 10 CFR 76.35 (a) USEC is required to submit, as part of its application for a certificate of compliance, a training program that meets the requirements of 10 CFR 76.95. Section 76.95 requires USEC to establish, implement, and maintain a training program "for individuals relied upon to operate, maintain or modify the GDPs in a safe manner." Section 76.95 also states that the training program for such individuals "shall be based on a systems approach to training that includes the following:

1. Systematic analysis of the jobs to be performed,
2. Learning objectives derived from the analysis that describe desired performance after training,
3. Training design and implementation based on the learning objectives,
4. Evaluation of trainee mastery of the objectives during training,
5. Evaluation and revision of the training based on the performance of trained personnel in the job setting.

This section describes the overall training program in place at PORTS, including the training applicable to personnel who are "relied upon to operate, maintain, or modify the GDPs in a safe manner." Personnel are trained to recognize and cope with safety hazards they may encounter in their jobs. They are also trained on practices important to public safety, safeguard of licensed material, and protection of the environment.

The overall training program at PORTS is comprised of several basic components. These components include:

- **General Employee Training** for persons who require unescorted access, including visitors (Section 6.6.4);
- **Operations and Maintenance Technical Training** for those persons relied upon to operate, maintain or modify structures, systems, and components (SSCs) on the Q-List (Section 6.6.5);
- **Radiation Worker Training** for those persons who require unescorted access to or perform work in restricted areas - this program is not SAT-based (Section 6.6.6);
- **Health Physics Technician Training** for those persons identified in the Radiation Protection portion of this application - this program is SAT-based (Section 6.6.7);
- **Emergency Management Training** for those persons identified in the Emergency Plan portions of this application - these programs are not SAT-based (Section 6.6.8.1);
- **Fire Protection** for those persons who are identified in the Fire Protection portion of this application - this program is not SAT-based (Section 6.6.8.2);

- **Environmental, Safety and Health (ES&H) Training** for those persons who have training requirements defined by laws and regulations - this program is not SAT-based (as defined in Section 6.6.9);

- **Subcontractor Training** for those temporary personnel performing maintenance or modifications to the GDPs - this program is not SAT-based (Section 6.6.10);

- **Nuclear Criticality Safety Engineer/Specialist Training** for those persons identified in the Nuclear Criticality Safety portion of this application - this program is SAT-based (Section 6.6.11);

- **Quality Control Inspection and Independent Audit Personnel Training** for those persons identified in the Quality Assurance Plan portion of this application - this program is not SAT-based (Section 6.6.12);

- **Manager Training** for those first and middle level managers who manage and supervise personnel "relied upon to operate, maintain or modify the GDPs in a safe manner" - this program is not SAT-based (Section 6.6.13);

- **Cascade Controller/Coordinator Training** for those personnel who direct the overall operations of the gaseous diffusion cascade - this program is SAT-based (Section 6.6.14);

- **Plant Shift Superintendent Training** for those persons who provide managerial oversight for the operation of the Plant Uranium Enrichment Facility and other support activities - this program is SAT-based (Section 6.6.15);

- **System Engineer Training** for those persons who review modifications to Q or AQ-NCS items - this program is SAT-based (Section 6.6.16);

- **Laboratory Technician Training** for those persons who work in the Laboratory Technician classification - this program is SAT-based (Section 6.6.17);

- **Environmental Technician Training** for those persons who work in the Environmental Technician classification - this program is not SAT-based (Section 6.6.18).

- **Security Education** required by 10 CFR 95.33 for persons granted access authorization clearances - this program is not SAT-based (Section 6.6.19); and

- **Utilities Operator Training (Utility Operator and Distribution & Inspection Operators)** for those persons who operate and monitor air, nitrogen, water, sewage, steam and cooling systems - this program is not SAT-based (Section 6.6.20).

- **Power Operator Training** for those persons who operate and monitor the plant electrical distribution system - this program is not SAT-based (Section 6.6.21).

6.6.1 Training Program Organization and Administration

The PORTS Training Organization consists of a centralized staff which reports directly to the Training Manager. The training staff consists of technical trainers, administrative personnel and mid-level managers who are directly responsible for assisting the plant in the design, development, implementation, and auditing of training programs in the following functional areas:

- General Employee Training,
- Operations and Maintenance Technical Training,
- Radiation Worker Training,
- Health Physics Technician Training,
- Environmental, Safety and Health Training,
- Subcontractor Training,
- Training Records, and
- Training Instructor/Developer Qualification.

Central Training staff personnel are assigned by the Training Manager to interface with functional line managers to coordinate training development and implementation for functional areas.

The Training Manager is responsible for establishing procedures that effect the development and implementation of training programs at PORTS. Formal training will be developed using the systems approach to training methodology for those tasks associated with the operation, maintenance or modification of structures, systems or components identified as Q or AQ-NCS items.

The Organization Managers are responsible for defining the job-specific training needs and ensuring completion of training and qualification for personnel within their organization. Workers relied upon to operate, maintain or modify Q or AQ-NCS items are trained and evaluated for qualifications prior to assignment of these duties. Task or duty area qualification is granted by line management based on successful evaluation of the worker's mastery of the learning objectives presented during training. Maintenance of qualification is contingent upon successful completion of continuing training and/or through satisfactory on-the-job training evaluations.

Line Management, i.e., Group and Organization Managers, develop and maintain a description of each organization's training requirements. These requirements are identified in Training Requirement Matrices (TRMs) approved by the Group/Organization Manager and the Training Manager. Training attendance is tracked by the training organization and line management based on these requirements. Training notifies line management of personnel who have not successfully completed initial training or who are past due for identified continuing training. Line management is responsible for placing work restrictions or removing employees from duty where training is deficient.

6.6.1.1 Initial Training

Initial training contains the classroom and OJT training necessary to provide an understanding of the fundamentals, basic principles, systems, procedures, and emergency responses involved in an employee's work assignments. Initial task or duty area qualification is granted by line management based on successful evaluation of the employee's mastery of the learning objectives presented during the training. Applicable process safety training (e.g., TSR training and nuclear criticality safety training) is included in the initial training.

TSR training is provided to facility operators and managers specific to their area of responsibility. It is designed to provide a sufficient understanding of the safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls necessary for the safe operation of the GDPs. TSR training is reviewed by Nuclear Regulatory Affairs as a subject matter expert. TSR refresher training is required every 2 years.

Nuclear Criticality Safety training based on ANSI/ANS-8.20-1991, "American National Standard for Nuclear Criticality Safety Training" is provided for personnel who handle or manage the handling of fissile material and work within Fissile Control Areas. This training is reviewed and approved by the Nuclear Criticality Safety Technical Staff and includes a discussion of the following:

- The fission process,
- Controllable factors and examples of their application at this facility,
- NCS postings, and
- Consequences of some of the historical criticality accidents.

Managers of personnel described above receive additional training on the managerial responsibilities relating to Nuclear Criticality Safety.

Personnel who work under procedures containing NCSA requirements receive additional training on these procedures with emphasis on NCS limits and controls. NCS refresher training is required every 2 years.

Reportable events are analyzed by appropriate plant personnel as described in SAR Section 6.9 to determine the root cause and contributing factors. Corrective actions involving training are assigned, scheduled and tracked to completion. Lessons learned come from many different sources and are shared with others quickly through plant bulletins, memoranda, and required reading files. Lessons learned which have an impact on initial training are factored into training materials prior to the delivery of the next training session.

6.6.1.2 Continuing Training

Continuing training is provided for employees in the interest of promoting safety, safeguards and security, and environmental protection awareness. Continuing training is also provided as a means to maintain and improve job-related knowledge and skills.

Plant procedures and Organization TRMs contain training requirements which delineate continuing training for employees. For those positions that are SAT Based, the Training Requirements are defined in each Training Development and Administrative Guide (TDAGs). The number of hours dedicated to this training annually is determined based on the following factors:

- Frequency required by regulatory agencies and national standards.
- Overtrain tasks identified in SAT-based programs.
- Training needs as determined by line management. This includes, but is not limited to, nuclear criticality safety assessments, facility system changes, component changes, procedure changes, lessons learned (including industry and in-house operating experiences, event reports), emergency response procedures, and TSRs.

The Operational Drill Program is used to augment continuing training. Drills are conducted to reinforce training and to assess the operational readiness of workers in the job environment.

6.6.2 Trainee Selection

Selection of personnel will be made in conformance with the established general employment policies. Minimum qualifications for each position are included in the employment policies. Additionally, employment will be contingent upon satisfactory completion of a physical examination, drug screening test, and the ability to obtain the required security clearance. For those job positions which are SAT-based, the minimum qualification requirements for selection into the training program are defined in each TDAG.

Personnel may be exempted from training as defined in plant training procedures. New hire or position incumbents will be considered for exemption from segments of classroom training and OJT. Exemptions will be based on one of the following methods:

1. Management review of an individual's prior training records and/or job performance history provides information stating that the individual has achieved the necessary required skills; or

2. Employee demonstrates minimum knowledge requirements by passing module examination in lieu of training (test-out); or

3. Employee demonstrates minimum skills/proficiency requirements by successfully completing task performance evaluations in lieu of OJT.

Exemptions will be documented.

6.6.3 Systems Approach to Training

The formal training program for those personnel who operate, maintain or modify Q or AQ-NCS items is based on a SAT methodology. This includes the following elements:

- Conduct of needs/job analysis and identification of tasks for training,
- Development of learning objectives,

- Development of lesson plans and training guides,
- Evaluation of trainee mastery of learning objectives, and
- Evaluation of the effectiveness of training.

6.6.3.1 Conduct of Needs/Job Analysis and Identification of Tasks for Training

A needs/job analysis is used to identify the tasks affecting worker or public safety, safeguards of regulated material, or protection of the environment as identified in the application, including the Technical Safety Requirements (TSRs). The analysis is conducted utilizing either job incumbent/manager written surveys or the table-top method with subject matter experts. The training programs for the following job positions/worker classifications are based on a needs/job analysis:

- Cascade Operators,
- Chemical Operators,
- Uranium Material Handlers,
- Electricians,
- Instrument and Electronic Mechanics,
- Maintenance Mechanics,
- System Engineers,
- Cascade Controllers/Coordinators,
- Plant Shift Superintendents,
- Health Physics Technicians,
- Laboratory Technicians, and
- Nuclear Criticality Safety Engineer/Specialist.

The facility-specific task list is developed for each of the above positions/classifications. The task lists are analyzed based on input from line management and subject matter experts, rating each task on degree of difficultly, importance of the task, and frequency of task performance. From this analysis, the tasks are selected for training based on their rating. The ratings are:

- **Overtrain** — requiring initial and continuing training,
- **Train** — requiring initial training,
- **Pre-train** or **just-in-time** — requiring training but will not be taught until that specific knowledge or skill is needed, or
- **No train** — formal training not required.

The tasks selected for training are matrixed to the associated procedures and training materials. The associated training materials are updated as necessitated by changes in procedures, facility system/equipment, or job scope prior to conducting training.

Tasks related to the on-site transportation of UF_6 are covered in the applicable SAT-based training programs.

Procedure changes, equipment changes, facility modifications and other changes affecting task performance, are monitored and evaluated for their impact on the development or modification of initial and continuing training programs. The affected training materials are modified or new materials developed, based on the significance of the change, and modifications are documented in the program files. Training is required prior to implementation of new or modified procedures except where line management (with the concurrence of training management) has determined that there are no changes to the SAR (including accompanying programs and plans), TSRs, or NCSAs; the task is easily performed; the task is performed frequently; and changes to the procedure do not affect the original intent of the procedure.

6.6.3.2 Development of Learning Objectives

Learning objectives are established to identify the training content and to define satisfactory trainee performance for the task or group of tasks selected for training. Learning objectives state the requisite knowledge, skills, and abilities the trainee must demonstrate. The conditions under which the required actions will take place and the standards of performance required of the trainee are also determined in development of the learning objectives. Learning objectives are sequenced within training materials based on their relationship to one another.

Learning objectives are documented in lesson plans and training guides and are revised as necessary based on changes in procedures, facility systems/equipment, or job scope.

6.6.3.3 Development of Lesson Plans and Training Guides

Learning objectives derived from the rated task lists are analyzed to determine the appropriate training setting. Classroom lesson plans, on-the-job training guides, or other instructional materials are procured or developed based on this instructional analysis and design. Lesson plans and other training guides provide the guidance and structure necessary to ensure consistent delivery of training material from trainer to trainer and class to class.

Classroom lessons are used primarily to provide cognitive learning on the fundamentals, theory, basic operating and maintenance principles, individual systems, system inter-relations, safety requirements, and processes used at PORTS.

Other forms of instructional materials, such as video, computer-based training and self-study may be used as alternatives or supplements to classroom instruction.

OJT is a systematic method of providing training on job-related skills and knowledge for a position. This training is conducted in the actual work environment and demonstrates actual task performance whenever practical and may be implemented on-shift if management determines that manpower and operational conditions will not be impacted by the training activities. Applicable tasks and related procedures for each technical area provide the input for the OJT that is designed to supplement and complement training received through formal classroom or laboratory training and to ensure personnel are qualified to perform their assigned tasks.

Classroom lesson plans, OJT guides, and other instructional materials receive technical reviews by designated subject matter experts and instructional reviews by training management before approval

and use. Training materials are approved by the responsible Organization or Group Manager and Training before issuance.

Designated subject matter experts or knowledgeable technical trainers provide classroom training and/or OJT evaluations. These personnel receive training and are qualified on the instructional methods and techniques applicable to the training setting.

Training Development and Administrative Guides (TDAGs) describe programs developed for a specific SAT-based job/position. This description includes:

- Organization and Administration Responsibilities,
- Trainee Selection Criteria,
- Course Loading For Initial and Continuing Training,
- Training Resource and Facility Guidelines,
- Test/Evaluation Guidelines,
- Training and Evaluation Documentation Guidelines, and
- Course/Modules for Specific Qualification Areas (listed by title and numbers).

6.6.3.4 Evaluation of Trainee Mastery of Learning Objectives

Within the job position/worker classification training programs are training courses. Each training course contains logical instructional blocks or "modules" presented in such a manner that specific learning objectives are accomplished. Trainee progress is evaluated by technical trainers and line management through a variety of performance demonstrations such as written examinations, oral examinations, and practical tests to ensure mastery of the job performance requirements or learning objectives contained in these modules. Remediation is provided as appropriate.

6.6.3.5 Evaluation of the Effectiveness of Training

Systematic evaluations of training effectiveness and its relation to on-the-job performance are used to ensure that the training program conveys required skills and knowledge and to revise the training, where necessary, based on the performance of trained personnel in the job setting. The student feedback of the training received and the first-line manager's evaluation of the student's performance on the job after training is completed are utilized to determine the training effectiveness and areas for refinement. Student feedback occurs at several points in the training program. At the completion of training, the student evaluates the instructor and course. Post training evaluations of the effectiveness of training is requested from students and supervisors three to six months after completion of training. Each of these evaluations is specified in plant training procedures.

Self-assessments and evaluations of the individual training programs are conducted by line and training management as described in Section 6.8.2. Nuclear Safety and Quality auditors provide additional assessments through the audit program. These assessments and evaluations are used to determine training program strengths and weaknesses.

6.6.3.6 Training Instructor/Developer Qualification

Training Instructor/Developer Qualification is the responsibility of PORTS Training. The Basic Instructor/Developer Qualification program is administered by the Training staff. Training is provided to designated subject matter experts (SMEs) and/or knowledgeable technical trainers who develop and/or conduct classroom training and/or OJT evaluations for site personnel. The program consists of several modules designed to train instructor/developers in the application of the SAT methodology. The program includes initial training and qualification and periodic re-evaluation of skills and knowledge in both material development and/or instructional competency. It is the responsibility of line management to identify plant personnel who are required to complete this program.

Employees selected as training instructors/developers for SAT-based employee training and qualification programs at the GDPs have work experience in their areas of training responsibility. Instructor/developers are selected by management based on identification of an individual as a Subject Matter Expert (SME) within the area they are assigned to develop or implement training programs. If instructors are not assigned offices in the work area of the plant, they are required to spend a minimum amount of time on a quarterly basis, working in the area they instruct. Instructors are expected to stay up to date on procedures/policy changes, modifications to TSRs, and lessons learned through the required reading program. Instructors and subcontractors hired to develop training materials have ready access to designated subject matter experts who assist them when developing training materials. All SAT-based training program materials must be reviewed and approved by subject matter experts and line management prior to implementation.

6.6.4 General Employee Training

General Employee Training (GET) is a biennial training requirement for all employees, subcontractors, and visitors who request, and who are required to have, unescorted plant access. GET includes the following subject areas:

- General Employee Radiological Safety,
- Nuclear Criticality Safety,
- General Topics,
- Hazard Communication, and
- Emergency Preparedness.

6.6.4.1 General Employee Radiological Training

General Employee Radiological Training (GERT) covers the employee's responsibilities for maintaining exposures to radiation and radioactive materials in accordance with the low as reasonably achievable (ALARA) philosophy. The training reviews natural background and manmade sources of radiation, the whole body radiation dose limit for nonradiological workers, the potential biological effects from chronic radiation doses, embryo and fetus protection, ALARA concepts and practices, and methods used to control radiological materials and contamination. If a person requires unescorted access to a restricted area, additional radiological safety training is provided as discussed in Section 6.6.6.

6.6.4.2 Nuclear Criticality Safety

An overview of the Nuclear Criticality Safety (NCS) program is provided. The training emphasizes the prevention of accidental nuclear criticality, describes the hazards and risks of a nuclear criticality accident, explains nuclear criticality safety responsibilities, and teaches the proper response to a nuclear criticality alarm.

Additional NCS training based on ANSI/ANS-8.20-1991, "American National Standard for Nuclear Criticality Safety Training" is provided for personnel who handle or manage the handling of fissile material and work within Fissile Control Areas. This training is described in Section 6.6.5.1.

6.6.4.3 General Topics

General Topics include a general overview of (1) health and safety awareness programs, (2) the employee's rights and responsibilities and the employer's duties as defined by laws and regulations, and (3) use of procedures and conduct of operations.

6.6.4.4 Hazard Communication

The purpose of this awareness-level training is to inform personnel that hazardous chemicals are present in the work place and to help them understand the function of warning labels and signs, Material Safety Data Sheets (MSDSs), and the written Hazard Communication Program.

Additional job-specific chemical safety training is provided to those personnel who handle or supervise the handling of hazardous chemicals identified in the chemical safety portion of this application.

6.6.4.5 Emergency Preparedness

This training introduces personnel to basic Emergency Plan elements including (1) emergency plan safety objectives and priorities, (2) ways to report emergencies, (3) recognition and correct responses to plant alarm signals, (4) evacuation guidelines for radiological and nonradiological emergencies, (5) personnel accountability procedures, and (6) personnel responsibilities during emergencies.

6.6.4.6 Visitor Site Access Orientation

Site access training for plant visitors who will be escorted consists of self study of an orientation handbook that covers the following general information:

- Driving Rules
- Compliance with postings and signs
- Use of eye, head, hearing, and respiratory protection
- Emergency Phone Numbers
- Radiological protection concerns

- Emergency Preparedness
- Security requirements and limitation of access and items prohibited

6.6.5 Operations and Maintenance Technical Training

Training and qualification programs for the following operations personnel are established, implemented, and maintained based on a systems approach to training. Programs are organized in parts and/or phases to support current management/organization structure and are subject to change. Initial and continuing training is provided for the following operations and maintenance job categories relied on to operate, maintain, modify, or test Q or AQ-NCS items:

- **Cascade Operators** — This program is designed for personnel who operate and monitor uranium enrichment process equipment and systems. Cascade Operators work in building control rooms, UF_6 withdrawal areas, and cell and operating floor levels of the process buildings and maintain process controls, execute valving orders, and conduct sampling.

  The Cascade Operator training program is separated into two phases with a variable duration depending on the operator's assignment and ability:

  Phase I provides classroom training on the core fundamentals required for all new operators and consists of the following: Cascade Operations Orientation, Uranium Enrichment Technology, Operating Principles of Gaseous Diffusion Equipment, Process Control, Process Support Systems, Power, and Process Building Utilities.

  Phase II provides classroom and on-the-job training leading to qualification on operating and building systems within the Cascade complex and consists of the following: General Cascade Operator Qualification, X-326 Unit & Utility Operator Qualification, X-326 ACR Operator Qualification, X-326 Withdrawal Operator Qualification, X-333 Unit & Utility Operator Qualification, X-333 ACR Operator Qualification, X-333 Withdrawal Operator Qualification, X-333 Cold Recovery Operator Qualification, X-330 Unit & Utility Operator Qualification, X-330 ACR Operator Qualification, X-330 Withdrawal Operator Qualification, X-330 Cold Recovery Operator Qualification, X-300 Plant Control Facility Qualification.

- **Chemical Operators** — This program is designed for personnel who perform decontamination and chemical cleaning services, operate UF_6 feed and fluorine generation systems, solution treatment and uranium recovery processes and waste handling operations. Chemical Operators work primarily in uranium enrichment process buildings, UF_6 feed and fluorine generation facilities, and in two chemical process cleaning/decontamination facilities.

The **Chemical Operator** training program is separated into two phases and has a variable duration depending on the operator's assignment and ability.

Phase I provides classroom training on the core fundamentals required for all Chemical Operators.

Phase II consists of job specific classroom and OJT training leading to task qualification in assigned duty areas. This includes job specific hazardous chemicals and nuclear criticality safety training. Duty areas within the Chemical Operator classification typically include Small Parts, Recovery, Tunnel, Field Decontamination, Microfiltration, Biodenitrification, Chemical Cleaning, UF_6 Feed and Fluorine Generation.

- **Uranium Material Handlers** — This program is designed for personnel who operate and monitor liquid UF_6 transfer and sampling systems, receive, transport and ship UF_6 cylinders, store and inventory UF_6 and non-UF_6 uranium bearing containers. Uranium Materials Handlers work in the Toll Enrichment Facility, UF_6 feed and sampling facilities, uranium enrichment process buildings, and in storage facilities.

The Uranium Material Handling program is divided into Parts I through IV as described below.

Part I, General provides training on the common tasks shared by all Uranium Material Handlers including: Nuclear Criticality Safety, Mobile Equipment Operation, Cylinder Inspection, Weighing Operations, Inventory, and Contamination Control.

Part II, Autoclaves provides system knowledge and duty area training on the operation of steam heated autoclaves, plus the heating, transfer, and sampling of UF_6 product material utilizing steam heated autoclaves, and Technical Safety Requirements.

Part III, X-344 Shipping & Receiving / Pick-Up & Delivery provides training in: the preparations, inspections and paperwork requirements involved in shipping and receiving of enriched UF_6 by motor carrier; the physical transferring and storage of uranium bearing containers.

Part IV, X-343 UMH Operations provides duty area training in the responsibilities of UMH personnel assigned to the X-343 building. Training includes sampling of feed and product cylinders utilizing X-343 autoclaves, and the preparations, inspections, and paperwork requirements involved in shipping and receiving feed cylinders of UF_6 by both truck and rail.

All new employees entering the UMH classification must successfully complete Parts I & II listed above to become qualified. As assignments dictate, employees receive training in Parts III & IV.

- **Electricians** — This program is designed for 1st Class (Journeyman) Electricians who service and repair electrical equipment and systems in buildings and switchyards plantwide.

The Electrician training program is described in phases below. The duration of the Electrician training program depends upon the new employee's "skill-of-the-craft," job assignment and individual ability.

Phase I provides general training for employees in the maintenance classification.

Phase II provides fundamentals training for Electricians.

Phase III provides classroom and/or OJT training leading to task qualification on the following systems and/or equipment: Autoclave Systems, UF_6 Liquid Handling Overhead Cranes, Smoke Detector Heads, Cascade Withdrawal Systems and Power Distribution Systems, Emergency Power Systems, Decontamination and Chemical Cleaning Systems, Building Alarms, Radiation Alarms, Fire Protection Systems, and UF_6 Pigtails. Personnel are trained to perform tasks in their assigned duty areas.

- **Instrument and Electronic Mechanics** — This program is designed for 1st Class (Journeyman) Instrument and Electronic Maintenance Mechanics who service and repair pneumatic and electronic instruments in facilities throughout the plant.

The Instrumentation & Controls (I&C) training program for Instrument and Electronic Maintenance Mechanics is described in phases below. The duration of the I&C training program depends on the new employee's "skill-of-the-craft," job assignment and individual ability.

Phase I provides general training for employees in the maintenance classifications.

Phase II provides fundamentals training for Instrument and Electronic Maintenance Mechanics.

Phase III provides classroom, laboratory and/or OJT training leading to task qualification on the following systems and/or equipment: Cascade Withdrawal Systems, Line Recorders, Space Recorders, Mass Spectrometers, FTIR Buggies, Microfiltration, Calciners, Plant Radiation Monitoring and Detection Instruments and Alarms, Autoclave Safety Systems, HF and F_2 Operating Systems, Radio Operated Cranes, selected Pressure/Temperature/Air Flow Transmitters, UF_6 Pigtail Fabrication, Nuclear Criticality Alarm Systems, Public Warning Systems, Digital Scales, CADP Outleakage Systems, and Fire Protection Systems. Personnel are trained to perform tasks in their assigned duty areas.

- **Maintenance Mechanics** – This program is designed for 1st Class (Journeyman) Maintenance Mechanics who install, remove, repair, and service plant equipment and systems in the field and in shop locations.

The training program for Maintenance Mechanics is described in phases below. The duration of the Mechanical Maintenance training program depends on the new employee's "skill-of-the-craft," job assignment and individual ability.

Phase I provides general training for employees in the maintenance classifications.

Phase II provides fundamentals training for Maintenance Mechanics.

Phase III provides classroom and/or OJT training leading to task qualification on the following systems and/or equipment: Cylinder Stacking; the installation and/or repair and preventative maintenance of Compressor Seals, Scales, selected Process Valves & Pumps, Autoclaves, and UF_6 Liquid Handling Overhead Cranes. Personnel are trained to perform tasks in their assigned duty areas.

Skill of the craft is established as a prerequisite for job selection. If an employee is classified as a journeyman, then that person has demonstrated competency of "skill-of-the-craft." If deficiencies in a journeyman's qualifications are revealed during job specific training and performance evaluations designed to measure the employee's mastery of learning objectives, then remedial training is provided on the appropriate craft fundamentals to maintain "skill-of-the-craft" expectations.

The training is designed, developed, and implemented to assist facility employees in gaining an understanding of applicable fundamentals, procedures, and practices specific to the gaseous diffusion process and facility. It is also used to develop the skills necessary to perform assigned work in a safe manner. If a task is identified to operate, maintain, or modify a specific Q system or process, then the training will be developed using SAT methodology. The training is categorized as Initial Training and Continuing Training.

6.6.6 Radiation Worker Training

Radiation Worker Training is a biennial training requirement for personnel whose job requires them to have unescorted access to radiological restricted areas. The training includes a comprehensive curriculum consisting of the following, as appropriate:

- Fundamentals of atomic structure, radiological definitions, types of ionizing radiation, units of measurement, dose, and dose rate calculations;
- Biological effects of ionizing radiation including cell sensitivity and chronic and acute exposure;
- Radiation work permit applications and use;
- Radiation limits for occupational and non occupational workers as well as the general public;
- ALARA practices for protection from exposure to radiation or radioactive materials;
- Personnel Monitoring Programs in place to monitor the worker's exposure to radiation;
- Radioactive Contamination Control to minimize and control the spread of contamination;
- Radiological Postings and Controls for familiarization with the signs and postings in the work area;
- Emergencies involving radiological material and the correct response; and
- Chemical Toxicity of Soluble Uranium Compounds.

This training includes knowledge examinations and practical factor examinations of the personal protective equipment, personnel monitoring, and radiation measurements, if needed. Radiation Worker

training is reviewed and approved by the Radiation Protection Manager and administered by the Training organization. The extent of the course material shall be commensurate with the potential for exposure.

6.6.7 Health Physics Technician Training and Qualification

Health Physics Technician Training and Qualification is administered by the training organization in accordance with guidelines provided in the TDAG for Health Physics Technicians. It utilizes the systems approach to training (Section 6.6.3) and applies to those individuals, both plant and contractor, who will be engaged in the evaluation of radiological conditions in the nuclear facilities and the implementation of the necessary radiological safety measures as they apply to nuclear facility workers and members of the general public.

6.6.8 Fire Protection and Emergency Management Training

6.6.8.1 Emergency Management Training

Emergency Management Training is administered by Emergency Management under the direction of the Plant Services Manager. It is defined in the Portsmouth Emergency Plan (Sections 7.2 and 7.3). Training is conducted in the areas of:

- General Emergency Plan training (Section 7.2.1)
- Specialized Emergency Plan training for the Emergency Response Organization (Section 7.2.2)
- Offsite Emergency Management training (Section 7.2.3)

Emergency Management drills and exercises are conducted to develop, maintain, and test the response capabilities of personnel, facilities, equipment, and training (Section 7.3).

6.6.8.2 Fire Protection Training

Fire Protection Training is administered by Plant Services and is covered in Section 5.4.5.

Fire Services personnel are trained and equipped to handle anticipated types of emergencies. Emergency medical response personnel meet requirements for state certification as emergency medical technician (these are usually also firefighters). Qualified instructors provide a range of classroom and hands-on training to maintain standards of performance for all response personnel. Training needs are reviewed annually and the training program modified to meet identified needs. State certification requirements provide the basis for firefighter training programs. Drills are conducted quarterly, as part of the plant emergency plan.

6.6.9 Environmental, Safety and Health Training

This training covers those environmental, worker safety and health subject areas required by applicable local, state and federal regulations and is provided to personnel commensurate with their job assignments. Specific modules identified as required compliance training for PORTS employees is contained in each organization's training requirement matrix. Some of the areas include:

- Radiological worker safety
- Nuclear Criticality Safety
- Respiratory training
- Confined space entry
- Asbestos worker safety
- Hearing conservation
- Temperature extremes
- OSHA Hazard communication
- Hoisting and Rigging
- Mobile Equipment Operaitons
- Lockout/Tagout Work Permits
- Safety and Health Work Permits
- RCRA for Hazardous Waste Generators
- OSHA HAZWOPER
- Fall protection
- Personal Safety (PPE)
- Lead Hazards

6.6.10 Contractor Training

Contractor training requirements are determined by the applicable site technical representative Training Group Manager. This determination will be based upon the site access requirements and job functions of each specific contract.

6.6.11 Nuclear Criticality Safety Engineer/Specialist Training

Nuclear criticality analyst training and qualification are administered by the Criticality Safety section. Training is based on ANSI/ANS-8.20-1991, "American National Standard for Nuclear Criticality Safety Training," and Criticality Safety procedures that define educational and experience prerequisites for incumbents, along with required training courses and OJT activities to be completed prior to qualification.

6.6.12 Quality Control Inspection and Independent Audit Personnel Training

The qualification and re-qualification of inspection personnel, auditors, lead auditors and nondestructive examination personnel is performed in accordance with QAP Section 2.2.4.

6.6.13 Manager Training

Manager Training is provided for those first and middle level managers who manage the operations and maintenance personnel relied upon to operate, maintain, or modify Q or AQ-NCS items. The training is designed, developed, and implemented to assist facility managers in gaining an understanding of the applicable procedures and practices specific to the gaseous diffusion process and facility. Also, it is used to develop the managerial and leadership skills necessary to effectively manage personnel.

6.6.14 Cascade Controller/Coordinator Training

Cascade Controller/Coordinator Training is administered by Operations and provided to those persons who direct the overall operations of the gaseous diffusion cascade. This training is based on the systems approach to training (Section 6.6.3) and is designed, developed, and implemented to provide the Cascade Controllers/Coordinators an understanding of the overall integration of the process and support systems necessary to operate the GDP. Cascade Controllers/Coordinators also receive Manager Training (Section 6.6.13).

6.6.15 Plant Shift Superintendent Training

Plant Shift Superintendent Training is administered by the Shift Operations Manager and provided to those persons who provide managerial oversight for the daily operations of the Plant Uranium Enrichment Facility and other support activities. This training is based on the systems approach to training (Section 6.6.3) and is designed, developed, and implemented to provide the Plant Shift Superintendent an understanding of the overall integration of the processes, support systems, administrative and emergency procedures, and regulatory reporting requirements necessary to operate the GDP. Plant Shift Superintendent qualification is granted by the Operations Manager upon successful completion of training. The Plant Shift Superintendent's training program is structured into several distinct but interrelated courses that include the following:

1. Incident Command and Emergency Response
2. Occurrence Reporting; Problem Identification; Evaluation, Disposition and Regulatory Notifications
3. Technical Subjects

6.6.16 System Engineer Training

System Engineer Training is administered by Engineering and is provided to those persons who provide engineering support and review of the modifications to Q or AQ-NCS items. System Engineers are responsible for reviewing design proposal and modifications; ensuring that the appropriate documents and procedures are updated to be consistent with modifications; and assisting in work control preparation and identification of post-maintenance test requirements for Q systems. The training is based on a detailed review of job analysis data, training requirements for specific systems, and existing training materials.

6.6.17 Laboratory Technician Training

Laboratory Technician Training is administered by Production Support in accordance with the guidelines set down in the Administrative Guide for the Laboratory and Technician Training Program. The training is based on a detailed review of job analysis data of procedure reviews. The analysis results were used to establish the learning objectives, test items, instructional methods, and instructional settings. Training is provided in the areas of Laboratory Controls & Standards, Mass Spectrometry, Process Services, Chemical Technology, Uranium Sampling, and Uranium Analysis.

6.6.18 Environmental Technician Training

Environmental Technician Training is administered by Production Support and is provided to those persons who operate, repair, calibrate, and troubleshoot environmental process facilities. Training is provided in the areas of handling and sampling hazardous liquid and solid wastes and other environmental media.

6.6.19 Security Education

Security Education briefings are administered by Plant Services as described in the Security Plan for the Protection of Classified Matter. The briefings are described as Initial/Comprehensive Briefings, Contractor Personnel Briefings, Termination Briefings, Foreign Travel Briefings, and Group Briefings. Security refresher briefings are coordinated by the security section and administered by plant training.

6.6.20 Utilities Operators (Utility Operator and Distribution & Inspection Operators)

This program is administered by the training organization and is designed for personnel who operate and monitor air, nitrogen, water, sewage, steam, and cooling systems. Operators work primarily in pump houses, water treatment facilities, the steam plant, uranium enrichment process buildings and cooling towers. They also perform inspections and valving operations on distribution piping for these systems plantwide. They receive formal training based on a job and needs analysis.

6.6.21 Power Operator Training

Power Operations Training is administered by the training organization and is designed for personnel who operate and monitor the plant electrical distribution system. Operators work primarily in the electrical switchyards and Plant Control Facility. They also perform inspections and switching operations for facilities plantwide.

6.6.22 Maintenance of Training Records

Training attendance records, examinations, employee qualification records, and program needs are maintained in an accurate, auditable manner to document each employee's training. The individual employee

training records are maintained in both hard copy and electronic files by PORTS Training. The electronic file system consists of a database relating the required training identified by the cognizant line manager to the actual training date of the employee. This electronic file is used for readily determining the status of an employee's qualification status. The training program records contain the Job Analysis data and history of the modules to include records of review and revisions.

6.6.23 Section Deleted

6.6.24 Training Programs on AQ Activities

Training for AQ activities will differ from training for Q activities in the rigor and formality for individual SAT elements, as described in the Quality Assurance Program, Appendix A.

6.7 HUMAN FACTORS

6.7.1 Description

Human factors are considered in the conduct of operations at PORTS. Those processes affecting the ability of personnel to carry out their assigned responsibilities incorporate guidance that provides assurance that human performance is considered, commensurate with the complexity, and safety impact of the activity being performed.

Human factors considerations are incorporated in several PORTS practices, including:

- Engineering design work associated with new equipment and facility modifications.
- Preparation, validation, and use of procedures.
- Development of training and qualification of personnel who operate, maintain, or modify structures, systems, and components (Q-SSC) relied upon for safety.

In addition, reports that identify human factors problems associated with existing facility design or procedures and training provide an information base to identify needs and improve human factors considerations.

Safety functions that require human actions in order to accomplish are considered as follows:

- Safety functions that require human actions are conducted in accordance with procedures, as described in Section 6.11. These procedures take into account the environmental, cognitive, and physical considerations that can affect the performance of the activity under consideration. Procedure writers' guides are used to ensure that the procedures clearly identify the required actions, and that necessary prerequisites, cautions, and warnings are incorporated in a manner that will give the performer adequate notice before proceeding with a procedural step that requires special consideration. The procedure writer's guide incorporates human factors guidance for writing procedures; these are considered during the procedure development/preparation "basic element" of the procedure process described in Section 6.11. The procedure development program that results from application of this guidance will ensure that these procedures will have had the appropriate human factors applied.

- The procedure development process incorporates a walk-through and/or demonstration step, which ensures that the procedure will result in the desired actions. In addition to the accuracy of the procedure, the ability of the performer to implement the steps of the procedure is assessed. Human factor considerations such as accessibility, visibility, ergonomic capability, suitability of environment for the required activity, and interferences are observed, and if required, action is taken to mitigate any adverse elements.

- Systematic approach to training addresses consideration of human factors requirements through the job task analysis process. This process considers each element of the task to be accomplished, and identifies those aspects for which development of training requirements is needed. The task analysis breakdown will reveal sequences of steps or requirements for supporting information or observation that are potentially problems from a human factors perspective. This will provide the information necessary to allow the correction of such deficiencies. Application of the job task analysis process is described further in Section 6.6.

- Verification that human factor deficiencies have not been incorporated in the procedure development or training is accomplished through conduct of drills and exercises, which provides a basis for evaluating operator performance required to accomplish safety functions. Observation, coupled with operator feedback during post-exercise critiques, provides indicators that identify any human factor deficiencies that may have been encountered.

- Event reporting and problem analysis also leads to identification of potential human factor deficiencies. Root cause analysis of events and performance problems occurring during operation applies a systematic process that includes consideration of human factors deficiencies as drivers. Any such deficiencies will be incorporated in the problem reporting and corrective action systems, ensuring they receive management attention on a systematic basis.

- The processes described above may identify physical factors that can impede expected quality of performance. Any changes in plant or component design recommended to improve human performance will be considered by the design and engineering organizations, and will be evaluated to determine their disposition. Engineering organizations are provided guidance in application of human factors elements to the design and specification process. Additionally, design requirements are established with input from the operators and the responsible system engineer. Elements that can impact performance, such as accessibility, reach, visibility, and layout are considered in the design wherever appropriate. Maintenance requirements, such as front panel or pull space location of test points, are considered, as well as operational needs.

- "Operator Aids" are used to provide assistance to operators in performing their duties (e.g., pin charts in the Area Control Rooms to identify the position of valves).

Human factor considerations and improvements can be adequately handled through existing management controls, specifically engineering design, procedures, training and problem analysis. Any changes in human actions required to prevent or mitigate accidents (specified in Chapter 2 of the TSRs) that may involve introduction of complex cognitive challenges will be specifically evaluated for human factor considerations as they are identified.

Additionally, human actions required by the TSRs to prevent or mitigate accidents (specified in Chapter 2 of the TSRs) are systematically evaluated for human factor considerations, including accessibility, visibility, ergonomic capability, suitability of the environment for the required activity, and interferences, on a 3-year cycle. Any improvements identified by the TSR evaluation process will be handled through the problem reporting system for tracking to resolution. The evaluation of human actions required by the TSRs to prevent or mitigate accidents is documented and used in the routine procedure reviews and in the evaluation of plant physical and programmatic changes. The evaluation does not include an evaluation of the existing plant and process controls design. There is no formal "Human Factors Review" program. The simplicity of the safety design of the gaseous diffusion plant does not require a formal "Human Factors Review" program.

6.7.2 Items Addressed by Compliance Plan

None.

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6.8 AUDITS AND ASSESSMENTS

A system of audits and assessments is implemented at the Portsmouth Gaseous Diffusion Plant (PORTS) to ensure that the health, safety, and environmental programs, as described in this application, are adequate and effectively implemented. The system is designed to ensure comprehensive program oversight at least once every three years. The system is comprised of two distinct levels of activities. These are audits and assessments.

6.8.1 Audits

Audits are conducted by the USEC plant independent Nuclear Safety and Quality organization in accordance with procedures and checklists by qualified auditors. Audits verify the effectiveness of health, safety and environmental programs and their implementation and determine the effectiveness of the assessment process.

These audits and their associated frequencies are conducted in accordance with Section 2.18 and Appendix A of the Quality Assurance Program and use written plans and checklists. Audits are performed under the direction of a Lead Auditor, qualified in accordance with the American Society of Mechanical Engineers (ASME) NQA-1 Supplement 2S-3. Lead auditors and staff auditors are functionally and organizationally independent of the programs and activities examined. Where appropriate, audit teams are supplemented with on site and/or off site technical specialists.

Audit results are documented and reported as specified in plant procedures. Provisions are made for immediate reporting and corrective action where warranted, as described in Section 6.9. A plant corrective action program is administered to ensure proper control of corrective actions as defined in Section 2.16 of the Quality Assurance Program (QAP).

6.8.2 Assessments

Assessments are performed by management responsible for implementing portions of the QAP to assess the adequacy of the part of the QAP for which they are responsible and to assure its effective implementation, as required by NQA-1 Basic Requirement 2. Personnel from the area being assessed may perform the assessment, provided that they do not have direct responsibility for the specific area being assessed (does not apply to Section 6.8.2.3). Results of assessments shall be documented. Any observations from the program assessments are resolved by the responsible organization manager.

Assessments performed include:

6.8.2.1 Program Assessments

The managers responsible for several major programs assure assessments are performed of their particular programs:

- Nuclear Criticality Safety activities related to assessments are addressed in Section 5.2.2.9, "Operation Surveillance and Assessments."

- Assessments are conducted of the fire protection program in Organizational Self Assessment (Section 6.8.2.2) and Management Self Assessments Activities (Section 6.8.2.3). These assessments include the Testing and Inspection program as addressed in Section 5.4.4 and the hazard evaluations of Section 5.4.1.2.

- Assessments of environmental programs are performed to assess compliance with environmental laws and regulations and to verify completion of corrective actions for environmental noncompliances.

- Assessments of health and safety programs are performed on a regular basis (typically one to two program areas per month, with all major program areas, as determined by the Organization Manager, assessed every 3 years) by Safety and Health professionals to verify that programs are effectively achieving their designed purposes. Industrial Safety, Radiation Safety and Chemical Safety are the major programs assessed.

- Assessments of the accuracy of nuclear materials accounting records are performed by Nuclear Material Control and Accountability (NMC&A) Group personnel. These assessments are required as part of the item monitoring activities associated with the item control program, as described in Section 8 of the Fundamental Nuclear Material Control Plan (FNMCP).

- Consistent with Section 2.2 of the QAP, an assessment of the status, adequacy, and effectiveness of the QAP at each GDP is provided to the USEC Vice President, Operations, at least once every 24 months by the NS&Q Manager at each GDP.

6.8.2.2 Organizational Self Assessments

Managers of Operations, Maintenance, Production Support, Engineering, Plant Services, and Training maintain a management assessment process within their organization to assess the adequacy of and effectiveness of the implementation of the programs under their cognizance. The Nuclear Safety and Quality organization will monitor the organizational self-assessment process.

6.8.2.3 Management Self Assessment Activities

Managers evaluate findings from audits, assessments and problem reports to determine their significance to plant safety and to ensure proper resolution. In addition, a program titled Management by Walking Around (MBWA) describes how senior managers evaluate plant facilities in the areas of Occupational Safety and Health, Nuclear Criticality Safety, Radiological Protection, Environmental Compliance, Safety Requirements, Conduct of Operations, and Conduct of Maintenance. Issues relating to Training, Quality Assurance, Maintenance, Configuration Management, etc., are also assessed during these assessments. Problem Reports (as described in Section 6.8.2.4) are generated for nonconforming conditions.

6.8.2.4 Problem Reporting

All plant employees have the responsibility to write problem reports on safety, operating and noncompliance items. Problem reports are screened routinely to assign an owner for each problem reported, as well as to determine if identified problems represent significant conditions adverse to quality (SCAQ) based on established criteria. A plant corrective action program is administered to ensure proper control of corrective actions as defined in Section 2.16 of the Quality Assurance Program (QAP). Corrective actions are tracked through completion.

6.8.3 Section Deleted

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6.9 EVENT INVESTIGATIONS AND REPORTING

USEC is required by 10 CFR 76.35(a)(7) to describe its management controls and oversight program governing activities directly relevant to nuclear safety and safeguards and security. This section describes the event investigations and reporting elements of the management controls and oversight program. USEC management ensures that activities directly relevant to nuclear safety and safeguards and security are conducted in a controlled manner to protect employees and public health and safety, the environment, and national security interests. As part of this program, abnormal events are identified, reported, and investigated. This includes identification and categorization of the event as well as an analysis to determine the root cause and its effect upon nuclear safety and safeguards and security. The corrective actions developed as a result of the investigation are entered into a commitment management database, which is a dedicated database used to track and trend corrective actions and plant commitments. This database also facilitates management's review of the corrective actions and plant commitments.

USEC is required by 10 CFR 76.120 and other applicable sections of the regulations referenced in 10 CFR 76.60 to notify the NRC of certain plant events and conditions and to determine the cause, corrective actions, and lessons learned. USEC shall satisfy these requirements by following administrative procedures relating to problem reporting (Problem Reporting Procedure) and nuclear regulatory event reporting (Nuclear Regulatory Event Reporting Procedure). Key aspects of the problem reporting and event reporting procedures are described in Section 6.8 and this section. These procedures work together to ensure that abnormal events and conditions occurring at the plant are promptly reported to appropriate plant personnel, assessed, and when required, reported to the NRC Operations Center or designated NRC office. Table 6.9-1 indicates those events for which a notification is required and those events for which a written report will be provided to the NRC.

6.9.1 Event Identification

In accordance with the Problem Reporting Procedure (PRP), plant personnel are required to report to their line manager or directly to the Plant Shift Superintendent (PSS) abnormal events or conditions that may have the potential to harm the safety, health, or security of on-site personnel, the general public, or the environment. The initial reporting to the PSS is as described in the PRP. Plant personnel are also required to immediately report conditions which may require emergency response.

6.9.2 Event Categorization

In accordance with the Nuclear Regulatory Event Reporting Procedure (NERP), the PSS assesses and categorizes abnormal events or conditions using the notification and reporting criteria set forth in 10 CFR 76 and the other applicable regulations referenced in 10 CFR 76.60. Table 6.9-1 provides a list of the initial event notification and reporting criteria. In making the assessment, the PSS may consult with plant management or other personnel possessing expertise or knowledge concerning the type of event or condition being assessed.

6.9.3 NRC Notification

If an event or condition within the USEC-leased area, or from a USEC directed activity outside the government reservation, is categorized as a reportable event, the PSS shall provide initial notification (verbal or written, as required) to the NRC Operations Center or designated NRC office and provide, to the extent known at the time of notification, the information specified in 10 CFR 76.120(d)(1) and other applicable sections of the regulations referenced in 10 CFR 76.60. Notification shall be made as soon as possible but not later than the time period(s) stated in the regulations. Reporting time periods are shown with the reporting criteria listed in Table 6.9-1. Verbal and/or written communication involving classified information shall be conducted in accordance with the Security Plan for the Protection of Classified Matter that is part of this application.

Site events that occur within the DOE managed property (USEC non-leased area) shall be categorized by the PSS. Events classified as an occurrence shall be reported to DOE by the PSS. These events shall be assessed by the PSS and other plant management, as needed, to determine whether further action is required to be taken by USEC personnel, including reports to the NRC.

6.9.4 Event Investigation

All reportable events, where a follow-up written report is required, are investigated to determine the root cause and corrective actions necessary to prevent recurrence. This investigation is conducted and documented, as required, by the plant corrective action process. Other events not requiring a follow-up written report are evaluated using the corrective action process to determine actions to be taken. The magnitude of this investigation is dependent upon the significance and complexity of the reportable event. This may range from the assignment of an event investigation team or may be as simple as a one-person investigation. Documentation related to the event investigation may be separate from the event report and will be retained in accordance with Records Management and Document Control requirements described in Section 6.10.

6.9.5 Written Report

When required by Table 6.9-1, a report summarizing the results of the event investigation is prepared in accordance with the NERP. The report shall contain, at a minimum, the information specified in 10 CFR 76.120(d)(2) or other applicable sections of the regulations referenced in 10 CFR 76.60. The written report shall be forwarded to the NRC within the time specified in Table 6.9-1 or applicable NRC regulations. The PORC is required to review each written event report prior to submission to the NRC.

6.9.6 Corrective Actions

For each reportable event, corrective actions to prevent recurrence shall be developed by responsible management. These actions shall be validated and tracked to ensure proper implementation and closure, in accordance with the plant corrective action process. Prior to closure, the actions taken to prevent recurrence shall be reviewed by management. Evidence files used to support action closure shall be maintained in accordance with approved records management procedures.

6.9.7 Lessons Learned

Lessons learned from observations, experiences, events, and problem reports are reviewed and effectively communicated across both plants. The Operating Experience Review Program (OERP) Administrator reviews gaseous diffusion operations and reviews related industry operating experience information to determine whether it warrants informational review or action by the organization with responsibility within the area of concern. Lessons learned from NRC-regulated facilities, as they apply to gaseous diffusion operations, shall be included.

6.9.8 Section Deleted

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Table 6.9-1. Event notification and reporting criteria applicable to USEC.

*Unless otherwise noted, notifications are verbal.
(W) = written (follow-up) report

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
A. Criticality Event		
1. After the discovery of a criticality event.	1 Hour	70.52(a) 76.120(a)(1)
	60 Day (W)	76.120(d)(2)
2. For operations that comply with the double contingency principle:	4 Hours From Initial Observation	NRC BL 91-01 Supp. 1
a. moderation is used as the primary criticality control, or		
b. more than a safe mass of fissionable material is involved (regardless of the type of controls used to satisfy the double contingency principle), and		
c. that meet one or more of the following:		
(1) Any event that results in the violation of the double contingency principle, as defined in ANSI 8.1, and where double contingency cannot be re-established within 4 hours after the initial observation of the event.		
(2) The occurrence of any unanticipated or unanalyzed event for which the safety significance of the event or corrective actions to re-establish double contingency are not readily identifiable.		
(3) Any case where it is determined that a criticality safety analysis was deficient and where the necessary controlled parameters were not established or maintained.		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
A. Criticality Event (continued)		
(4) Any event involving a controlled parameter previously identified by the NRC or the certificate holder as requiring immediate reporting to the NRC and where double contingency cannot be re-established within 4 hours after the initial observation of the event.		
3. For operations that do not comply with the double contingency principle, for which:	4 Hours From Initial Observation	NRC BL 91-01, Supp. 1
a. moderation is used as the primary criticality control, or		
b. more than a safe mass of fissionable material is involved, and		
c. that meet one or more of the following:		
(1) The occurrence of any unanticipated or unanalyzed event for which the safety significance of the event or corrective actions to re-establish the approved controls are not readily identifiable.		
(2) Any case where it is determined that a criticality safety analysis was deficient and where the necessary controlled parameter is not established or maintained.		
(3) Any event involving the controlled parameter and the approved control on the parameter cannot be re-established within 4 hours after initial observation of the event.		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
A. Criticality Event (continued)		
4. All other criticality safety events:	24 Hours	NRC BL 91-01, Supp. 1
a. For violations involving operations that comply with the double contingency principle and do not meet the aforementioned criteria (listed above), but still result in a violation of the double contingency principle, such as events where the double contingency principle is violated but control is immediately re-established.		
b. For violations involving operations that do not comply with the double contingency principle and do not meet the aforementioned criteria (listed above), but still result in a violation of a control relied on for the criticality safety of the operation.		
B. Special Nuclear Material Event		
1. Any loss, other than normal operating loss, of special nuclear material (SNM).	1 Hour	70.52(a),(b) 74.11(a) 76.120(a)(2)
	60 Day (W)	76.120(d)(2)
2. Any theft or unlawful diversion of SNM that USEC is authorized to possess, or any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion of SNM.	1 Hour	70.52 (b) 74.11(a) 76.120(a)(3)
	60 Day (W)	76.120(d)(2)
3. Discovery of any unauthorized production of enriched uranium.	1 Hour	74.11(a)
4. Discovery of the loss of any shipment of SNM or spent fuel.	1 Hour	73.67(g)(3)(iii) 73.71(a)(1)
	30 Day (W)	73.71(a)(4)
5. Recovery of or accounting for each lost shipment.	1 Hour 30 Day (W)	73.71(a)(1) 73.71(a)(4)

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

	Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
	B. Special Nuclear Material Event (continued)		
6.	Significant supplemental information (about the loss or recovery or accounting of any lost shipment) that becomes available after initial telephonic notification or after submission of the written report.	Report As Discovered (within 1 hour) 30 Day (W)	73.71(a)(5)
	C. Theft or Loss of Licensed Material (Refer to Appendix C of 10 CFR 20 for Quantities)		
1.	Any lost, stolen, or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in Appendix C to 10 CFR 20 under such circumstances that it appears to USEC that an exposure to persons in unrestricted areas could result.	Immediately After Known (within 1 hour) 30 Day (W)	20.2201(a)(1)(i) 20.2201(b)(1)
2.	Within 30 days after the occurrence of any lost, stolen, or missing licensed material becomes known to USEC, all licensed material in a quantity greater than 10 times the quantity specified in Appendix C to 10 CFR 20 that is still missing at this time.	30 Day 30 Day (W) After making verbal notification	20.2201(a)(1)(ii) 20.2201(b)(1)
3.	Any additional substantive information on the loss or theft of licensed material.	30 Day (W) After Learning of Such Information	20.2201(d)
	D. Emergency Conditions (Refer to Plant Specific Emergency Action Levels)		
1.	An emergency condition that has been declared an Alert.	1 Hour 60 Day (W)	76.120(a)(4) 76.120(d)(2)
2.	An emergency condition that has been declared a Site Area emergency.	1 Hour 60 Day (W)	76.120(a)(4) 76.120(d)(2)

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
E. Protective Action Prevented		
An event (events may include fires, explosions, radiological releases, etc.) that prevents immediate protective actions necessary to avoid releases, or exposures to radiation or radioactive materials that could exceed regulatory limits.	4 Hours 60 Day (W)	76.120(b) 76.120(d)(2)
F. Fire Explosion Damaging Radioactive Material (Refer to Appendix B of 10 CFR 20)		
A fire or explosion damaging any radioactive material or any device, container, or equipment containing radioactive material when: 1. the quantity of material involved is greater than five times the lowest annual limit on intake specified in Appendix B to Sections 20.1001-20.2402 of 10 CFR 20 for the material; and 2. the damage affects the integrity of the radioactive material or its container.	24 Hours 60 Day (W)	76.120(c)(4) 76.120(d)(2)
G. Exposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Limits		
1. Any event involving byproduct, source, or SNM possessed by USEC that may have caused or threatens to cause any of the following conditions:		
a. An individual to receive:	Immediate (within 1 hour)	20.2202(a)(1)
(i) A total effective dose equivalent (TEDE) of 25 rems (0.25 Sv) or more; or		
(ii) A lens dose equivalent of 75 rems (0.75 Sv) or more; or	30 Day (W)	20.2203(a)(1)
(iii) A shallow-dose equivalent to the skin or extremities of 250 rads (2.5 Gy) or more; or		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
G. Exposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Limits (continued)		
b. The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot cells or process enclosures).	Immediate (within 1 hour) 30 Day (W)	20.2202(a)(2) 20.2203(a)(1)
2. Any event involving loss of control of licensed material possessed by USEC that may have caused, or threatens to cause, any of the following conditions:		
a. An individual to receive, in a period of 24 hours-	24 Hours 30 Day (W)	20.2202(b)(1) 20.2203(a)(1)
(i) A TEDE exceeding 5 rems (0.05 Sv); or		
(ii) A lens dose equivalent exceeding 15 rems (0.15 Sv); or		
(iii) A shallow-dose equivalent to the skin or extremities exceeding 50 rems (0.5 Sv); or		
b. The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).	24 Hours 30 Day (W)	20.2202(b)(2) 20.2203(a)(1)

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
G. Exposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Limits (continued)		
3. Doses in excess of any of the following:	30 Days Written Notification Report	20.2203(a)(2)
a. The occupational dose limits for adults in Section 20.1201; or		
b. The occupational dose limits for a minor in Section 20.1207; or		
c. The limits for an embryo/fetus of a declared pregnant woman in Section 20.1208; or		
d. The limits for an individual member of the public in Section 20.1301; or		
e. Any applicable limit in the license; or		
f. Air emissions of radioactive material to the environment excluding Radon-222 and its daughters such that the total effective dose equivalent that an individual member of the public likely to receive the highest dose would be in excess of 10 mrem per year (as determined by the Annual Air Emission Calculation completed to demonstrate compliance with 10 CFR 20.1302).		20.1101(d)
4. Levels of radiation or concentrations of radioactive material in	30 Days Written Notification Report	20.2203(a)(3)
a. A restricted area in excess of any applicable limit in the certificate; or		
b. An unrestricted area in excess of 10 times any applicable limit set forth in this part or in the certificate (whether or not involving exposure of any individual in excess of the limits in Section 20.1301); or		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
G. Exposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Limits (continued)		
5. USEC, being subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR Part 190 (Subpart B-Environmental Standards for the Uranium Fuel Cycle-Standards for Normal Operations), shall report to NRC levels of radiation or releases of radioactive material in excess of those standards, or of certificate conditions related to those sta ndards.	30 Days Written Notification Report	20.2203(a)(4)
6. USEC shall submit a written report of any planned special exposure (PSE) conducted in accordance with Section 20.1206.	Within 30 Days of Conducting PSE	20.2204 20.1206(f) 19.13(d)
H. Unplanned Contamination (Refer to Appendix B of 10 CFR 20 for Annual Limits on Intake)		
An unplanned contamination event that:		
1. Requires access to the contaminated area, by workers or the public, to be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry into the area;	24 Hours 60 Day (W)	76.120(c)(1)(i) 76.120(d)(2)
2. Involves a quantity of material greater than five times the lowest annual limit on intake specified in Appendix B to Sections 20.1001-20.2402 of 10 CFR 20 for the material; and		76.120(c)(1)(ii) 76.120(d)(2)
3. Causes access to the contaminated area to be restricted for any reason other than to allow isotopes with a half-life of less than 24 hours to decay prior to decontamination.		76.120(c)(1)(iii) 76.120(d)(2)

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
I. Unplanned Medical Treatment of Individual with Radioactive Contamination		
An event that requires unplanned medical treatment at a medical facility, excluding plant medical facility, of an individual with radioactive contamination on the individual's clothing or body.	24 Hours 60 Day (W)	76.120(c)(3) 76.120(d)(2)
J. Safety Equipment Failure/Actuations		
1. An event in which equipment is disabled or fails to function as designed when:		
a. The equipment is required by a TSR to prevent releases, prevent exposures to radiation and radioactive materials exceeding specified limits, mitigate the consequences of an accident, or restore this facility to a preestablished safe condition after an accident;	24 Hours 60 Day (W)	76.120(c)(2)(i) 76.120(d)(2)
b. The equipment is required by a TSR to be available and operable and either should have been operating or should have operated on demand; and		76.120(c)(2)(ii) 76.120(d)(2)
c. No redundant equipment is available and operable to perform the required safety function.		76.120(c)(2)(iii) 76.120(d)(2)

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
J. Safety Equipment Failure/Actuations (continued)		
2. USEC shall notify the NRC within 24 hours of any automatic or manual actuation of a Q safety system that results from an event or condition that has the potential for significant impact on the health or safety of personnel. Events having the potential for significant impact are those events where actual plant conditions existed that the system was designed to protect against. The intent of this reporting requirement is to ensure that the NRC is notified of those events where a Q safety system was actuated, either manually or automatically, in response to a valid signal. Per discussions with NRC staff, this reporting requirement specifically excludes the reporting of: A. Actuations which result from and are part of a pre-planned sequence during testing or operation; B. The actuation is invalid and: (1) Occurs while the system is properly removed from service; (2) Occurs after the safety function has already been completed; C. Actuations caused by invalid signals (e.g., non-safety system signal, instrument drift, spurious signals, human error, or other invalid signals). Invalid Q safety system actuations are documented and evaluated through the Problem Reporting System.	24 Hours 60 Day (W)	
K. Safeguards Events		
1. USEC, being subject to the provisions of Section 73.67 shall notify the NRC Operations Center within 1 hour of discovery of the safeguards events described in paragraph I(a)(1) of Appendix G to this part [10 CFR 73].	1 Hour 30 Day (W)	73.71(b)(1) 73.71(b)(2)

Table 6.9-1 Event Notification and Reporting Criteria Applicable to USEC (Continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
K. Safeguards Events (Continued)		
2. Significant supplemental information that becomes available after initial telephonic notification to the NRC Operations Center or after the submission of the written report must be telephonically reported to the NRC Operations Center and also submitted in a revised written report.	As Soon As Discovered (within 1 hour) 30 Day (W)	73.71(a)(5) [criterion] 73.71(b)(2) [reporting requirement] 73.71(a)(5)
L. Violations Regarding National Security Information/Restricted Data		
1. When a request for security facility clearance is to be withdrawn or canceled, the NRC Division of Facilities and Security will be notified by the requestor immediately by telephone so that processing for this approval may be terminated.	Immediately (Within 1 hour; normal working hours only) Promptly Written Confirmation	95.21 95.21

Table 6.9-1 Event Notification and Reporting Criteria Applicable to USEC (Continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
L. Violations Regarding National Security Information/Restricted Data (Continued)		
2. Any alleged or suspected violation of the Atomic Energy Act, Espionage Act, or other Federal statutes related to classified information (e.g., deliberate disclosure of classified information to persons not authorized to receive it, theft of classified information.)	Immediately (Within 1 hour with written confirmation within 30 days) *	95.57(a)
3. Any infractions, losses, compromises, or possible compromise of classified information or classified documents not falling within 10 CFR 95.57(a). A copy of the security incident log shall be submitted to NRC Region III and the NRC Division of Facilities and Security on a monthly basis.	Log incident within 24 hours of discovery	95.57(b)
M. Material Handling		
1. Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87(i); or	Immediately (within 1 hour)	20.1906(d)(1)

* Notifications made to NRC Regional Administrator and NRC Division of Facilities and Security (during normal NRC working hours) or; NRC Operations Center (after normal NRC working hours).

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
M. Material Handling (continued)		
2. External radiation levels exceed the limits of 10 CFR 71.47.	Immediately (within 1 hour)	20.1906(d)(2)
N. Transportation Incidents		
1. Each certificate holder who transports licensed material outside of the confines of its plant or other place of use, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the regulations appropriate to the mode of transport of Department of Transportation (DOT) in 49 CFR 170 through 189.	Earliest Practicable Moment (within 1 hour) 30 day (W)	71.5(a)(1)(v) 71.5(b) [requirement to notify NRC]
a. USEC shall particularly note DOT regulations in the following areas: ...(v) Accident reporting-49 CFR 171.15 and 171.16 49 CFR 171.15 Immediate Notice of Certain Hazardous Materials Incidents 49 CFR 171.16 Detailed Hazardous Materials Incident Reports		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
N. Transportation Incidents (continued)		
2. USEC shall notify the NRC of information identified by USEC as having for the regulated activity a significant implication for public health and safety or common defense and security. USEC violates this paragraph only if USEC fails to notify the NRC of information that USEC has identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within 2 working days of identifying the information. This requirement is not applicable to information already required to be provided to the NRC by other reporting or updating requirements.	2 Working Days	71.6a(b)
3. USEC shall report:	30 days	71.95
a. any instance in which there is significant reduction in the effectiveness of any authorized packaging during use; and		
b. details of any defects with safety significance in the packaging after first use; with the means employed to repair the defects and prevent their recurrence.		
c. Instances in which the conditions of approval in the Certificate of Compliance were not observed in making shipment.		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
O. Failure to Comply or Existence of a Defect		
1. USEC has adopted appropriate procedures to:	Written Notification 30 Days after Receipt of Information	21.21(a)(3) [criteria] 21.21(d)(3)(ii) [reporting requirement]
- Ensure that a director or responsible officer subject to 10 CFR 21 is informed as soon as practicable, and in all cases, within the 5 working days after completion of the evaluation described in 10 CFR Sections 21.21(a)(1) or 21.21(a)(2) if the construction or operation of a facility or activity:		
a. Fails to comply with the Atomic Energy Act of 1954 as amended, or any applicable rule, regulation, order, or license of the NRC relating to a substantial safety hazard, or		
b. Contains a defect.		
2. A director or responsible officer subject to 10 CFR 21 or a person designated under 10 CFR Section 21.21(d)(5) must notify the NRC when he or she obtains information reasonably indicating a failure to comply or a defect affecting:	2 Days via Facsimile	21.21(d)(1) (criteria) 21.21(d)(3)(i) (reporting requirement)
a. The construction or operation of a facility or an activity within the United States that is subject to the licensing requirements under 10 CFR 30, 40, 50, 60, 61, 70, 71 or 72 and that is within his or her organization's responsibility; or		
b. A basic component that is within his or her organization's responsibility and is supplied for a facility or an activity within the United States that is subject to the licensing requirements under 10 CFR 30, 40, 50, 60, 61, 70, 71, or 72.		

Table 6.9-1. Event notification and reporting criteria applicable to USEC. (continued)

Criteria	Notification/ Reporting Time	Reporting Requirement Reference (10 CFR Section)
P. Miscellaneous		
USEC shall notify the NRC of any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials.	4 Hours	

6.10 RECORDS MANAGEMENT AND DOCUMENT CONTROL

Introduction

Records Management and Document Control programs are established to ensure records and documents required by the QAP are appropriately managed and controlled. These programs are designed to meet the specific recordkeeping and document control requirements set forth in 10 CFR 76 and the applicable provisions of other parts of 10 CFR. These programs provide administrative controls that establish standard methods and requirements for collecting, maintaining, and disposing of records. These programs also ensure that documents are controlled and distributed in accordance with identified written requirements and authorizations. The administrative controls for the generation and revision of records and documents are contained in plant implementing procedures. The principal elements of each of the Records Management and Document Control programs and a brief description of the manner in which the functions associated with each element are performed are provided below, along with a list of the types of records that are retained for the duration of the NRC Certification of Compliance for the plant.

6.10.1 Records Management Program

The Records Management program provides direction for the handling, transmittal, storage, and retrievability of records. Records media may include microfilm, electronic (magnetic or optical), or hard copy. Records are categorized and handled in accordance with their relative importance to safety and storage needs. Special provisions are made for handling contaminated records and ensuring their inclusion in the program. Responsibility for the administration of the Records Management program rests with the Plant Services Manager. Responsibility for Records Management program compliance rests with the group managers generating records. This program is implemented through procedures that provide guidance for the following program elements.

6.10.1.1 Legibility, Accuracy, and Completeness

Documents designated to become records shall be legible, accurate, complete, and contain an appropriate level of detail commensurate with the work being performed and the information required for that type of record.

6.10.1.2 Identification of Items and Activities

Records clearly and specifically identify the items or activities to which they apply.

6.10.1.3 Authentication

Records are authenticated or validated by the Organization Manager of the organization which originates the record, or his designee, as specified in the procedure which controls the generation and revision of these records. This is in the form of a signature and date applied to the record.

6.10.1.4 Indexing and Filing

Methods are specified for indexing, filing, and locating records within the record system to ensure the records can be retrieved in a timely manner.

6.10.1.5 Retention and Disposition

Records retention times are specified in a retention schedule. The process for disposition of records that have reached the end of their retention lifetime is specified by procedures and conforms to applicable requirements.

6.10.1.6 Corrections

Corrections to records are approved by the organization which created the record unless other organizations are specifically designated. Changes are made by clearly indicating the correction, the date of the correction and the identification of the individual making the correction.

6.10.1.7 Protection of Records

Controls are established for protection of records from deterioration, loss, damage, theft, tampering, and/or unauthorized access for the life of the record. Requirements include instructions on protection of records by the record originator until they are transferred to Records Management. Instructions for the protection of special record media such as radiographs, photographs, negatives, microform and magnetic media are provided to prevent damage from excessive light, stacking; electromagnetic fields, temperature, humidity, or any other condition adverse to the preservation of those records. Records which cannot be duplicated are stored in a fashion that minimizes deterioration.

6.10.1.8 Storage Requirements

Records are stored in authorized facilities or containers providing protection from fire hazards, natural disasters, environmental conditions, infestations of insects, mold, or rodents. Storage facilities are maintained to ensure continuous protection of the records. Requirements are specified for both permanent and temporary storage of records.

Permanent Storage

Records are permanently stored in facilities satisfying the following requirements:

1. Storage in 2-hour-rated containers meeting National Fire Protection Association (NFPA) 232-1986 or NFPA 232 AM-1986 or both as clarified in Chapter 1, Appendix A, or

2. Storage of duplicate copies in separate facilities that are sufficiently remote from each other to eliminate the possibility of exposure to simultaneous hazards, or

3. Storage in facilities that have the following: doors, structures, frames, and hardware that comply

with a minimum 2-hour fire rating; a fire protection system; 2-hour fire rated dampers on all boundary penetrations; sealed floor surface to minimize concrete dust; adequate access and aisle ways; and a prohibition on eating, drinking, or smoking and performing work other than that associated with records storage or retrieval.

Temporary Storage

The Records Management and Document Control process requires that those completed records documenting nuclear safety or safeguards and security matters that are being held temporarily by originating organizations be properly protected by maintaining them in 1-hour, fire-rated containers. If 1-hour, fire-rated containers are used, they either bear an Underwriters Laboratory (UL) label (or equivalent) certifying 1-hour fire protection, or the containers are certified for 1-hour fire protection by an authorized individual competent in the field of fire protection. Procedural requirements are used to limit the length of time during which records may be maintained in temporary storage, based on the significance of the record.

6.10.1.9 Receipt of Records

A record transmittal process is used to formally transmit records to Records Management. The process includes a receipt acknowledgment that notifies the sending organization that the records have been received and accepted.

6.10.1.10 Access to Records and Accountability for Removed Records

Requirements for controlling access to records and maintaining accountability for records are provided to ensure that only authorized personnel have access to records and to prevent loss, damage, or inadvertent destruction of records.

6.10.1.11 Records Requirements for Procured Goods or Services

Records management requirements for goods or services procured from outside suppliers are specified in the applicable procurement documents. These requirements cover:

1. Supplier methods for collection, storage, and maintenance of records;
2. Identification of required records and applicable retention periods;
3. Records submittal plans or indexes;
4. Availability, accessibility, and if applicable, disposition criteria for records retained by the supplier; and
5. Accessibility of the supplier's records prior to the final transfer to the purchaser.

6.10.1.12 Control of Sensitive Records

Control, accountability, protection, and dispositioning of classified, Unclassified Controlled Nuclear Information (UCNI), and sensitive records are in accordance with the Security Plan for the Protection of Classified Matter, Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance and any other security and privacy requirements.

6.10.1.13 Types of Records

Records series which will be included in the Records Management program include, but are not limited to:

1. Transportation and shipping records for nuclear materials.
2. Radiation Protection records, including ALARA findings and occupational radiation exposure records.
3. Training, qualification, and requalification records.
4. Procurement documents/records.
5. Design documents and changes thereto involving design modifications made to safety systems and equipment.
6. Manuals, instructions, and procedures.
7. Certification documents.
8. Reportable event records.
9. Gaseous and liquid radioactive and hazardous waste records.
10. Safety evaluations and NCS analysis records.
11. Plant radiation surveys and environmental survey records (including radioactive releases).
12. PORC records.
13. QA activity records required by the QA program.
14. Regulatory agency reports and responses.
15. Safeguards and security records.

Specific records will be retained for a period of time specified by applicable NRC, Federal or State regulations.

6.10.1.14 Usage and Control of Computer Codes and Data

Computer programs used in the Records Management program are controlled and maintained in accordance with the "Computing and Telecommunications Security Manual" and Information Systems procedures. These requirements and practices provide for virus protection as well as access control to the Records Management program database and ensure continuing usability of the codes as hardware and software technology change. Routine backups of the Records Management database are performed by Corporate Information Technology (CIT) application administrators. Precautions are taken to ensure that computer data that constitute a record are stored in a format that is readily retrievable even as hardware and software technology evolve. The storage format of computer data is reviewed as required to determine threats to future retrievability, and if necessary, the data are translated to an updated format and verified acceptable.

6.10.1.15 Assessment

The overall effectiveness of the Records Management program is evaluated through the audit program described in Section 2.18 of the Quality Assurance Program (QAP). Deficiencies identified are corrected in a timely manner in accordance with the PRP.

6.10.2 Document Control Program

The Document Control program provides direction for the handling, distribution, and transmittal of documents important to nuclear safety and safeguards and security that specify quality requirements or prescribe activities affecting quality, such as procedures, drawings, and calculations. This program is implemented through procedures that provide guidance on the following program elements.

6.10.2.1 Unique Identifier

A unique identification number is assigned or obtained by the generator for each document requiring controlled distribution. Document Control concurs with the numbering scheme for each document type.

6.10.2.2 Approval and Release of Documents

For documents and changes to documents required by the Quality Assurance Plan (QAP), requirements are established for approval and release of those documents for distribution. Documents requiring review and approval by the Plant Operations Review Committee (PORC) are identified in the TSR for PORC. Requirements for the review and approval of procedures are identified in the TSR for Procedures and SAR Section 6.11. Controlled documents are approved by the organization authorized to approve them as identified in the procedures which control their generation and revision. Changes to controlled documents are approved and released by the organization that performed the document's initial approval unless other organizations are specifically designated. After approval, the documents are forwarded to Document Control for control and distribution to the personnel on the approved distribution list.

6.10.2.3 Master Copy

A master copy of all approved controlled documents is maintained by Document Control to ensure the document is available for controlled copy issuance.

6.10.2.4 Controlled Document Index and Distribution Lists

Creation and maintenance of a controlled document index and controlled distribution list(s) for each document or document type are required. The controlled document index is used to maintain a list of controlled documents and to track the current (latest) approved revision levels of those documents. The index is available to users to verify current document revision levels. The controlled document index and the distribution lists are maintained and updated by Document Control.

6.10.2.5 Copies of Controlled Documents

Each controlled copy is stamped, marked or otherwise identified. A method is established in procedures for duplicating and marking controlled documents so that duplicates are distinguishable from the controlled version. Copies of controlled documents that are not marked or otherwise identified in accordance with procedural requirements are considered information only.

6.10.2.6 Distribution

Controlled documents are distributed in accordance with controlled distribution lists to ensure that controlled documents are available in a timely manner at locations where work is being performed. Specific time requirements are established for controlled document distribution and receipt acknowledgment. Document Control uses a transmittal form to distribute controlled documents to copyholders. Copyholders sign, date, and return the transmittal form to confirm that they have received the documents. Document Control tracks the issuance and receipt of transmittals.

6.10.2.7 Voided, Canceled, or Superseded Documents

When notified by the generator of a controlled document that the document has been voided, canceled, or superseded, Document Control removes the document from distribution and notifies copyholders of the changed status.

The approved revised document is distributed at the time that the original document is superseded. The RMDC database is updated to identify the latest approved revision of the document. Distribution of revised documents is described in the Document Control Program procedure and using a Transmittal Form distributed by either interoffice mail or hand delivery. The holder of the Controlled Copy is required to acknowledge receipt by returning a signed Transmittal Form to Document Control. Document distribution is completed in accordance with the safety significance of the document being distributed.

6.10.2.8 Marking Sensitive Documents

Proper marking and handling of documents designated as classified, UCNI, or sensitive documents is accomplished in accordance with the "Security Plan for the Protection of Classified Matter," Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance and any other security and privacy requirements.

6.10.2.9 Change Documents

Change documents are documents which are used to modify controlled documents. Controls are also applied to the change documents to provide revision approval and distribution controls equivalent to the original document until completion of installation, at which time the original document is revised. Documents showing the current configuration are not changed until the modifications are completed.

6.10.2.10 Revision Identification

The controlled document revision level is clearly identified on the document.

6.10.2.11 Document User Responsibilities

Responsibilities of the end user and copyholders are defined. Responsibilities include requirements for the use of controlled documents and working copies. Copyholders of controlled documents update their controlled documents each time a revision or change is sent out, and promptly return the transmittal form acknowledging receipt.

6.10.2.12 Usage and Control of Computer Codes and Data

Computer programs used in the Document Control program are controlled and maintained in accordance with the "Computing and Telecommunications Security Manual" and Information Systems procedures. These requirements provide for virus protection as well as access control to the Document Control program database and ensure continuing usability of the codes as hardware and software technology change. Routine backups of the Document Control database are performed by CIT application administrators.

6.10.2.13 Assessment

The overall effectiveness of the Document Control program is evaluated through the audit program described in Section 2.18 of the Quality Assurance Program (QAP). Deficiencies identified are corrected in a timely manner in accordance with the PRP.

6.10.2.14 Archiving Documents

The record copy of all revisions of controlled documents are transmitted to Records Management in accordance with the requirements of the Records Management program.

6.10.3 Organization and Administration

The Plant Services Manager is responsible for the implementation of the Records Management and Document Control programs for the life of the certification at PORTS.

6.10.3.1 Responsibilities

The Plant Services Manager is responsible for:

- Directing all activities and personnel of the Records Management and Document Control programs.
- Directing the development, implementation, and maintenance of methods and procedures encompassing a records management program.

- Directing the development, implementation, and maintenance of methods and procedures encompassing a document control program.
- Assuring that all laws, codes, standards, regulations, and company procedures pertaining to recordkeeping and document control requirements are met.

6.10.3.2 Training and Qualifications

The Records Management and Document Control group manager requires a minimum of five years experience as a manager. No specific experience related to the control of documents or management of records is required, although previous technical or records management and document control experience is recommended.

6.10.4 Employee Training

General training in Records Management and Document Control is provided to employees as part of the general topics covered in General Employee Training (GET), as described in Section 6.6.

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6.11 PROCEDURES

In accordance with 10 CFR 76.35 (a)(7), USEC is required to describe its management controls and oversight program governing activities directly relevant to nuclear safety, safeguards, and security. This section describes that management controls program for the development, issuance, and control of procedures. Procedures that are not related to nuclear safety, safeguards, and security and do not involve or impact the plant or plant operation as described in the Certification Application or Compliance Plan are not governed by the requirements of this section.

6.11.1 Scope

In accordance with 10 CFR 76.35 (a)(7), 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. The procedures program will only follow the guidance contained in the specific subsections of ANS 3.2-1994 that are noted in Appendix B along with applicable exceptions/clarifications for each subsection. USEC's commitment to these specific subsections of ANS 3.2-1994 applies only to those provisions which deal with procedures programs and not quality assurance. USEC's quality assurance commitments are described in the Quality Assurance Program description.

A balanced combination of written guidance, craftsman skills, and work site supervision help achieve the quality workmanship essential to realize the goal to conduct nuclear safety, safeguards, and security activities for the protection of the public, plant employees, and the environment. A graded approach in the review and approval of procedures is utilized to provide the necessary rigor for safe plant operation, assure the company's commitments to meeting regulations and standards are maintained, and assure a balance of effective safety with practical efficiency in plant operations.

Procedures are intended to prescribe those essential actions or steps needed to safely and consistently perform operations and maintenance activities. These elements are outlined in a procedures management writers guide, and described in implementing procedures.

6.11.2 Procedure Hierarchy

The procedure hierarchy is established in four levels. The levels are:

- Level 1 - Policy statements issued by executive management that apply to Gaseous Diffusion Plant (GDP) personnel.
- Level 2 - Standard Practice Procedures issued jointly at multiple sites (including headquarters) or an individual site or headquarter's procedure that applies to more than one organization.
- Level 3 - Procedures issued at the organization level that apply to more than one group within a larger group or specific organization.
- Level 4 - Procedures issued within a group or subfunction.

6.11.3 Procedure Types

The following types of procedures are used by USEC:

- Administrative Procedures: Those procedures that deal with policy or programs/administrative systems, provide programmatic requirements and do not normally involve manipulation of equipment.

- Operating (non-administrative) Procedures: Procedures that direct or cause operation/maintenance of equipment or may directly affect any physical characteristics of equipment.

- Alarm Response Procedures (ARPs): Procedures that provide information that identifies the symptoms of the alarm, possible causes, automatic actions, the immediate operator action to be taken, and the required supplementary actions.

- Off-normal procedures: Procedures that describe actions to be taken during unusual or out-of-the ordinary situtions.

- Emergency Operating Procedures: Procedures directing actions necessary to mitigate potential events or events in progress that involve needed protection of on-site personnel, the public health, and safety, and the environment.

6.11.4 Procedure Process

Procedures are developed or modified through a formal process incorporating the change controls described in Section 6.3. The procedure process utilizes nine basic elements to accomplish procedure development, review, approval, and control. These elements are Identification, Development, Verification, Review and Comment Resolution, Approval, Validation, Issuance, Change Control, and Periodic Review. These elements are discussed in detail in the following paragraphs.

6.11.4.1 Identification

As a minimum, a procedure is required for any task that is described in the SAR, TSRs, Quality Assurance Program, Emergency Plan, Environmental Compliance Status and Environmental Monitoring Report, Fundamental Nuclear Materials Control Plan, Physical Security Plan for the Transportation of Special Nuclear Material of Low Strategic Significance, Physical Security Plan for the Protection of Special Nuclear Material of Low Strategic Significance, Security Plan for the Protection of Classified Matter, Radioactive Waste Management Plan, Depleted Uranium Management Plan, Decommissioning Funding Program Description, Supplemental Environmental Information Related to Compliance Plan or the Plan for Achieving Compliance with NRC Regulations at the Portsmouth Gaseous Diffusion Plant. Maintenance activities can be addressed by written procedures, documented work instructions, or drawings appropriate to the circumstances as discussed in Section 2.5 of the Quality Assurance Program, SAR Section 6.4.7, and Appendix A.6, paragraph (a), of ANS 3.2-1994.

Organization managers have the responsibility for identifying which tasks will be proceduralized within their areas of control, as required by their being identified by the criteria in the preceding

paragraph and listed in Appendix A to this section. Procedures are required for operator actions necessary to prevent or mitigate the consequences of accidents described in SAR Chapter 4.

Additionally, new or revised NRC certification requirements are evaluated to determine impact on existing implementing procedures or to identify the need for new implementing procedures. Procedures are reviewed following unusual incidents to determine if changes are appropriate based on the root cause and corrective action determination for the particular incident. Procedure changes that are necessary as a result of a system modification are addressed in Section 6.3 as part of the modification control process.

A procedure is normally not needed if the work is not complex or only involves a few actions (unless failure to properly conduct those actions could result in significant consequences), if the task requires those skills normally possessed by a qualified person (otherwise known as "skill-of-the-craft" as discussed in Section 2.5.3 of the Quality Assurance Program), or if the consequences of error are minimal. This decision can only be made if a procedure does not meet any of the criteria presented in the first two paragraphs of this section.

6.11.4.2 Development

Procedure development/preparation and quality is the user organization's responsibility. Procedure development is accomplished in accordance with procedure guidance. A general description follows:

- A system is in place to track and document the procedure process.
- The approval process for the procedure is as described in Section 6.11.4.5.
- The procedure use category is determined. This determination documents the designation of a procedure as In Hand (Continuous Use), General Intent (Reference Use), or Information Use. The designation is based on the administrative or non-administrative use of the procedure, and the safety or financial consequences of failing to adhere to procedural requirements. Procedure use is discussed in more detail in Section 6.11.5.
- Input and review by affected parties. Other selected reviews will be obtained, such as Quality Assurance to ensure that quality assurance requirements are identified and included in operating procedures.
- Interviews with procedure users and process walkdowns are utilized to ensure procedures are usable, reflect as-built conditions, process operations, and maintain management controls for nuclear safety, safeguards, and security. During development, regulatory commitments, TSR, SAR, QAP, and NCSA requirements are identified and noted as commitments in the procedure.
- As the procedure is drafted, attributes that enhance procedural use are included, such as standard style organization, format, cautions, and warnings.

6.11.4.3 Verification

Verification is a process that ensures the technical accuracy of the procedure and that it can be performed as written. Non administrative procedures are verified by the procedure owner/user during the procedure development/change process. There are two basic attributes of the verification process. The first attribute relates to the technical accuracy of the procedure. It ensures that all technical information including formulas, set points, and acceptance criteria are correctly identified in the procedure. The second attribute is administrative, in that it verifies the procedure format and style and that it is consistent with the procedure writing guide. Verification consists of a walk-down of the procedure in the field or a table-top walk through. A standard checklist is used to ensure required attributes are included.

6.11.4.4 Review

Draft new procedures and procedure changes are distributed for technical reviews and cross-discipline reviews, as needed.

Functional area and cross-discipline reviews are performed by individuals not having direct responsibility for processing the new procedure or procedure change. Comments/questions generated during the review process are resolved with the originating organizations. If comments are so extensive that resolution of the comments changes the intent of the original draft, the revised draft procedure is verified a second time, and the validation checked. 10 CFR 76.68 and intent/nonintent screenings are performed for new and changed procedures (except minor administrative changes that are processed according to the procedure process).

Reviews by plant personnel ensure that the operating limits and controls identified in the SAR and TSRs, as well as quality assurance, programmatic, and regulatory requirements, are specified in procedures.

6.11.4.5 Approval

Following the resolution of review comments, procedures are approved. Approval authority rests with the responsible manager. In addition, procedures requiring PORC review are:

1. Each new procedure required by Section 6.11.4.1.

2. Each proposed change to procedures required by Section 6.11.4.1 if:

    a. the proposed change requires a written safety analysis in accordance with 10 CFR 76.68, or

b. the proposed change results in a change to the documents listed in the first paragraph of Section 6.11.4.1, or

c. the proposed change constitutes an intent change (i.e., a change in scope, method or acceptance criteria that has safety significance).

New procedures or procedure changes that do not meet these criteria do not require PORC review and can be approved by the responsible manager.

Managers ensure that necessary training is completed prior to procedure implementation (see Section 6.6).

6.11.4.6 Validation

The purpose of procedure validation is to ensure that no technical errors or human factor issues were inadvertently introduced during the procedure review process. Validation is required for all new procedures or procedure changes that require PORC review. Validation is performed by qualified personnel and may be accomplished by detailed scrutiny of the procedure as part of a walk through exercise or as part of a walk through drill (particularly for emergency or off-normal procedures). If the particular system or process is not available for a walk through validation, talk through may be performed in the particular shop or training environment. Performance of procedure validation is documented.

6.11.4.7 Issuance and Distribution

Procedures are issued and controlled in accordance with the Records Management and Document Control program practices as described in Section 6.10.

6.11.4.8 Temporary Changes

Temporary changes to procedures required by Section 6.11.4.1 can be made provided:

1. the temporary change does not require a written safety analysis in accordance with 10 CFR 76.68, and

2. the temporary change does not result in a change to the documents listed in the first paragraph of Section 6.11.4.1, and

3. the temporary change does not constitute an intent change (i.e., a change in scope, method or acceptance criteria that has safety significance), and

4. the change is approved by two members of the plant management staff, at least one of whom is the Plant Shift Superintendent, and

5. the change is documented and reviewed in accordance with SAR Section 6.11.4 within 14 days of implementation.

Temporary changes to procedures may be made permanent once the change is reviewed and approved as required by SAR Section 6.11.4.

6.11.4.9 Periodic Review

Approved procedures are periodically reviewed to ensure their continued accuracy and usefulness. Procedures are periodically reviewed according to established criteria. The periodicity of these reviews is based on procedure content as follows:

Periodic Review Cycle	Procedures to Be Reviewed
1 year	Emergency Operating, Alarm Response and procedures dealing with highly hazardous chemicals as defined by the Process Safety Management (PSM) program.
3 years	Procedures not included as part of the 1-year review cycle that (1) are designated as In-Hand or (2) involve liquid UF_6 handling activities, off-normal procedures, and Nuclear Material Control and Accountability procedures.
5 years	Procedures not included as part of the 1-year or 3-year review cycles for (1) any tasks that are described in, or implement a commitment that is described in, the Certification Application (Volumes 1 through 4) or Compliance Plan; (2) operator actions necessary to prevent or mitigate the consequences of accidents described in Chapter 4; or (3) those activities listed in Appendix A.

Procedures not included as part of the 1-year, 3-year, or 5-year review cycle are periodically reviewed as deemed necessary by the responsible manager.

When conducting the periodic review, the procedure owner/SME performs a complete administrative and technical (requirements and references) review ensuring information is complete and accurate and that the procedure is usable as written.

6.11.5 Use and Control of Procedures

In-Hand (continuous use) procedures are performed step-by-step without deviation unless deviation is allowed by the procedure. General Intent (Reference Use) procedures are followed as written, unless deviation is allowed by the procedure. Information Use procedures are followed to implement programmatic requirements.

Controlled copies of procedures are marked "Controlled Copy". Working copies of procedures are marked "Working Copy," verified as the latest version, initialed, and dated prior to use. Information Only copies of In-Hand (Continuous Use) or General Intent (Reference Use) procedures are marked "Information Only" to indicate they are not controlled copies and are not used to perform work.

Work is stopped, the system is immediately placed in a safe condition, and corrective actions initiated if a step of a procedure cannot be performed as written, in accordance with site procedures.

6.11.6 Temporary Procedures

Temporary procedures may be issued only when permanent procedures do not exist (1) to direct operations during testing, maintenance, and modifications; (2) to provide guidance in unusual situations not within the scope of permanent procedures; and (3) to ensure orderly and uniform operations for short periods when the plant, a system, or component of a system is performing in a manner not covered by existing permanent procedures, or has been modified or extended in such a manner that portions of existing procedures do not apply. These temporary procedures may be used for a period of time which should not exceed 60 days, or a period for which the temporary condition must exist, whichever is greater. Temporary procedures that need to exceed the 60 days are assessed to ensure it is appropriate to extend the use of the temporary procedure. Completed assessments are reviewed by the PORC and are documented to provide technical justification for extending the use of the procedure for an additional period of time not to exceed 60 days. These temporary procedures are subject to the same level of review and approval as required for permanent procedures.

6.11.7 Records

Records generated during procedure use are identified in the governing procedure and controlled according to the plant Records Management and Document Control program practices as described in Section 6.10.

6.11.8 Items Addressed by Compliance Plan

This section is implemented as described with exception(s) as listed below. The listing of the exception(s) also contains a brief description of what is currently in place at the plant. The Compliance Plan provides a description of the exceptions (noncompliances), a justification for continued operation, a description of the actions to be taken to achieve compliance and the schedule for completion of those actions.

6.11.8.1 Section Deleted

6.11.8.2 Section Deleted

6.11.8.3 Section Deleted

6.11.8.4 PORC Review of Procedures

The PORC will review all procedures designated as In-Hand and procedures that involve liquid UF_6 handling activities within a 5-year period after the initial Certificate of Compliance is issued. This commitment only pertains to those procedures which will not otherwise be reviewed by the PORC (as required by Section 6.11.4.1), or by a PORC subcommittee, before the expiration of the 5-year period. Procedures in this scope have been, and will continue to be, reviewed by a PORC subcommittee, thereby satisfying this commitment for those specific procedures.

Appendix A

Activities defined by SAR Section 6.11.4.1 are the minimum activities that shall be covered by written procedures. In addition, any activity described in SAR Section 6.11.4.1 and listed below shall be covered by a written procedure (except for the maintenance activities listed below which may be covered by written procedures, documented work instructions, or drawings appropriate to the circumstances). This list is not intended to be all inclusive, since many other activities carried out during gaseous diffusion plant operations will be covered by procedures not included in this list. Similarly, this listing is not intended to imply that procedures be developed with the same titles as those in the list. This listing shall provide guidance on topics to be covered rather than specific procedures.

ADMINISTRATIVE PROCEDURES:

Training
Internal audits and inspections
Investigations and reporting
Records management and document control
Changes in facilities and equipment
Modification design control
Security and Visitor control
Quality assurance
Equipment control (lockout/tagout)
Shift turnover
Control room activities
Communications
Work control
Management control
Temporary modification
Procedures management
Nuclear criticality safety
Fire protection
Radiation protection
Radioactive waste management
Maintenance
Environmental protection
Packaging and transportation of nuclear material
Safety analysis
Chemical safety
Operations
TSR surveillances
Calibration control
Code inspections
Preventive maintenance

Appendix A (continued)

SYSTEM PROCEDURES THAT ADDRESS STARTUP, OPERATION & SHUTDOWN:

Cascade cells

Coolant
Freezer/sublimers
Purge cascade
Electrical power

Ventilation
Datum
UF_6 leak detection

Criticality alarms
Mass spectrometers
Cell leakrates
Cell negatives

Cascade sampling

Cell treatments
Deposit monitoring
Equipment removal with deposits
Cascade shift routines and operating practices
Reduction of cascade power level
Splitting and remaking the cascade
Cylinder burping
Product withdrawal operations
UF_6 cylinder filling
UF_6 cylinder handling

Liquid UF_6 handling crane operation

UF_6 autoclave operations, feeding, heating, sampling & transfer

UF_6 material handling equipment
Decontamination operations
Uranium recovery

Electrical switching operations
Electrical equipment inspections
Plant air
Plant nitrogen
Recirculating cooling water

Appendix A (continued)

Sanitary water
Plant water
Chemical trapping
Cold/controlled feeding

ABNORMAL OPERATION/ALARM RESPONSE:

Debladed compressor
Cell inleakage
Loss of cooling
Loss of instrument air
Loss of electrical power
Loss of autoclave containment
Loss of criticality alarm system
Cell load alarm
Cell coolant alarm
Fires

MAINTENANCE ACTIVITIES THAT ADDRESS SYSTEM REPAIR, CALIBRATION, INSPECTION, TESTING:

Repair of UF_6 smoke detectors
Repair of UF_6 valves
Testing of criticality alarm units
Testing of cranes
Compressor seal replacement
Calibration of autoclave control systems
UF_6 cylinder valve replacement
Calibration of process indicating instruments
Chemical trap changeout
Rigging
HEPA filter maintenance
Safety system relief valve replacement
Replacement of compressor motors
Replacement of autoclave gaskets
UF_6 cylinder inspection & testing
Liquid UF_6 handling crane inspection & testing

EMERGENCY PROCEDURES:

Response to a criticality
Toxic chemical releases (including UF_6)

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Appendix B

ANS 3.2-1994 Section/Paragraph Number	USEC Exception/Clarification
5.1	USEC utilizes only the requirements of this subsection which apply to procedures, and does not commit to those quality assurance aspects described in this subsection or any other subsection of ANS 3.2-1994.
5.2	USEC utilizes the requirements of this section as described in SAR Section 6.11.
5.2.2	USEC utilizes the requirements of this section with the exception that changes shall be made in accordance with the requirements contained in Section 6.11 and TSR 3.9, as appropriate, and not Technical Specifications, the exception that the requirements contained in 10 CFR 50.54(x)[2] do not apply to the gaseous diffusion plants (GDPs), the exception that "reactor operators" are "operators" at the GDPs, and the exception that the examples in ANS 3.2-1994 Subsections 5.3.7.1 and 5.3.7.2 do not apply to the GDPs.
5.2.3	USEC utilizes the requirements of this section, with the clarification that each GDP is not considered a multi-unit site.
5.2.4	No exceptions/clarifications.
5.2.5	USEC utilizes the requirements of this section with the exception of "refueling" (which does not apply to the GDPs), and the requirement to prescribe the review and approval process in the Technical Specifications. USEC prescribes the review and approval process in Section 6.11 and TSR 3.9, as appropriate.
5.2.18/1	USEC commits only to the administrative controls, and not quality assurance program aspects described in the paragraph or the rest of ANS 3.2-1994. The administrative controls only apply to activities affecting items described in the Quality Assurance Program (QAP).
5.2.18/2	The USEC commitments relative to human factors are fully contained in Section 6.7. The reference to a review of procedures six months after the first refueling outage does not apply to the GDPs. Procedures will be reviewed after unusual incident as required by a determination of root cause and corrective action as described in Section 6.9.
5.2.18/3	This paragraph does not apply to the GDPs.

Appendix B

ANS 3.2-1994 Section/Paragraph Number	USEC Exception/Clarification
5.2.18/4	The GDPs do not utilize technical specifications, therefore the second sentence in this paragraph does not apply to the GDPs. The frequency of periodic reviews is specified in Section 6.11.4.9.
5.2.18/5	No exceptions/clarifications.
5.2.18/6	No exceptions/clarifications.
5.2.18/7	USEC commitments relative to document control are fully contained in Section 6.10.
5.2.18/8	No exceptions/clarifications.
5.2.18/9	No exceptions/clarifications.
5.2.18/10	USEC commitments relative to document control are fully contained in Section 6.10.
5.2.18/11	USEC commitments relative to document control are fully contained in Section 6.10.
5.3	USEC utilizes the requirements of this section with the exception that requirements for procedure review and approval are not performed in accordance with Technical Specifications, but in accordance with Section 6.11 and TSR 3.9, as appropriate.
5.3.1	No exceptions/clarifications.
5.3.2	No exceptions/clarifications.
5.3.3	No exceptions/clarifications.
5.3.4	USEC utilizes the requirements of this section, with the clarification that USEC commitments relative to human factors considerations are fully contained in Section 6.7.
5.3.4.1	No exceptions/clarifications.
5.3.4.2	No exceptions/clarifications.
5.3.4.3	No exceptions/clarifications.
5.3.4.4	No exceptions/clarifications.

Appendix B

ANS 3.2-1994 Section/Paragraph Number	USEC Exception/Clarification
5.3.4.5	No exceptions/clarifications.
5.3.4.6	No exceptions/clarifications.
5.3.4.7	No exceptions/clarifications.
5.3.4.8	No exceptions/clarifications.
5.3.4.9	No exceptions/clarifications.
5.3.4.10	No exceptions/clarifications.
5.3.5	No exceptions/clarifications.
5.3.6	No exceptions/clarifications.
5.3.8.1	USEC utilizes only the requirements of this subsection which apply to procedures. Note that maintenance activities are described in Section 6.4.
5.3.8.2	No exceptions/clarifications.
5.3.8.3	No exceptions/clarifications.
Appendix A.6, paragraph (a)	No exceptions/clarifications.

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