December 14, 2000

Mr. J. M. Brown
Vice President - Operations
United States Enrichment Corporation
Two Democracy Center
6903 Rockledge Drive
Bethesda, MD 20817

SUBJECT: PORTSMOUTH INSPECTION REPORT 70-7002/2000011(DNMS)
AND NOTICE OF VIOLATION

Dear Mr. Brown:

On November 20, 2000, the NRC completed a routine resident inspection at your Portsmouth Gaseous Diffusion Plant. The purpose of the inspection was to determine whether activities authorized by the certificate were conducted safely and in accordance with NRC requirements. At the conclusion of the inspection, the inspectors discussed the findings with members of your staff.

Areas examined during the 6-week inspection period are identified in the report. Within these areas, the inspection consisted of a selective examination of procedures and representative records, interviews with personnel, and observations of activities in progress.

Based on the results of the inspection, the NRC has determined that one violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding the violation are described in detail in the enclosed report. The violation is of concern because of a continued lack of rigor in your staff's adherence to plant procedures. We will continue to monitor the effectiveness of your actions being taken to address the weaknesses, including those documented in your correspondence to us dated November 21, 2000.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned, and the date when full compliance will be achieved is already adequately addressed in the enclosed inspection report. Therefore, you are not required to respond to this violation unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed notice.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning these observations.

Sincerely,
/RA/
Patrick L. Hiland, Chief
Fuel Cycle Branch

Docket No. 70-7002
Certificate No. GDP-2

Enclosures:
1. Notice of Violation
2. Inspection Report 70-7002/2000011(DNMS)

cc w/encls:
P. D. Musser, Portsmouth General Manager
P. J. Miner, Manager, Regulatory Affairs/Commitment Management, Portsmouth
H. Pulley, Paducah General Manager
S. A. Toelle, Director, Nuclear Regulatory Affairs, USEC
Portsmouth Resident Inspector Office
Paducah Resident Inspector Office
R. M. DeVault, Regulatory Oversight Manager, DOE
S. J. Robinson, Portsmouth Site Manager, DOE
J. R. Williams, State Liaison Officer

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NOTICE OF VIOLATION

United States Enrichment Corporation
Portsmouth Gaseous Diffusion Plant

Docket No. 70-7002
Certificate No. GDP-2

During an NRC inspection conducted from October 10, 2000, through November 20, 2000, one violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, Revision 1, the violation is listed below.

Technical Safety Requirement 3.9.1 requires, in part, that written procedures shall be prepared, reviewed, approved, and implemented for activities described in Appendix A to Safety Analysis Report Section 6.11.

Safety Analysis Report Section 6.11, Appendix A, describes document control, procedures management, and equipment control as activities for which procedures shall be implemented.

In addition:

A. Procedure XP2-US-FO1109, "COP-9 Control Of Operator Aids," describes the requirements for review, approval, logging, and tracking operator aids.

B. Step 6.2.3 of Procedure UE2-PS-PS1034, "Use of Procedures," requires that "in-hand" procedures be taken to the work location, each step be read before performing it, and each step be performed in order unless deviation is allowed by the procedure.

C. Step 7.2 of Procedure XP2-GP-GP1033, "Lifted Leads And Jumpers," requires that leads that were lifted be recorded on the lifted lead and jumper sheet.

Contrary to the above:

A. On November 2, 2000, the inspectors identified several examples of failure by operations staff to implement the requirements of Procedure XP2-US-FO1109 in that operator aids were not properly reviewed, approved, and tracked.

B. On November 2, 2000, staff executing "in-hand" Procedure XP3-GS-MM4070, "Shaft Alignment," did not have the procedure at the work location and, therefore, did not read each step before performing it as required by Procedure UE2-PS-PS1034. In addition, steps were not performed in order as required, as data was not recorded in the procedure but was instead recorded on scratch paper and later transferred to the procedure.

C. On November 7, 2000, the inspectors identified that maintenance staff did not use a lifted lead and jumper log as required by XP2-GP-GP1033 during replacement of pressure transducer PT-705 in Building X-343.

This is a Severity Level IV Violation (Supplement VI). **(70-7002/2000011-01)**

The NRC has concluded that information regarding the reasons for Violation 70-7002/2000011-01, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance will be achieved are already adequately addressed in this Inspection Report. Therefore, a specific response to Violation 70-7002/2000011-01 is not required. However, you are required to submit a written statement or explanation, pursuant to 10 CFR 76.70, if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at Portsmouth, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response, with the basis for denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001.

Because your response will be placed in the NRC Public Electronic Reading Room (PERR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PERR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (for example, explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 14th day of December 2000

U.S. NUCLEAR REGULATORY COMMISSION
REGION III

Docket No:	70-7002
Certificate No:	GDP-2
Report No:	70-7002/2000011(DNMS)
Facility Operator:	United States Enrichment Corporation
Facility Name:	Portsmouth Gaseous Diffusion Plant
Location:	3930 U.S. Route 23 South P.O. Box 628 Piketon, OH 45661
Dates:	October 10, 2000, through November 20, 2000
Inspectors:	D. J. Hartland, Senior Resident Inspector J. M. Jacobson, Paducah Resident Inspector W. C. Gleaves, Mechanical Systems Engineer D. E. Roth, Fuel Cycle Inspector F. H. Burrows, Electrical Engineer
Approved By:	Patrick L. Hiland, Chief Fuel Cycle Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

United States Enrichment Corporation
Portsmouth Gaseous Diffusion Plant
NRC Inspection Report 70-7002/2000011(DNMS)

Operations

The inspectors identified several deficiencies regarding the use of operator aids by plant staff and a weakness with the required annual self-assessment in that it was not rigorous enough to identify the deficiencies. Plant staff took appropriate corrective action to address the issues. (Section O1.1)

Maintenance

Work packages reviewed were detailed and contained appropriate work instructions, procedures, and post-maintenance testing requirements. However, violations for failing to use a lifted lead and jumper log and failing to follow the "in-hand" requirements during execution of a procedure were identified. Plant management took appropriate action to address the issues. (Section M1.1)

The inspectors identified a smokehead that was not returned to its proper position after being moved to support a maintenance activity. In response, plant staff initiated appropriate action to ensure that an operability determination was performed prior to repositioning smokeheads in the future and that smokeheads were returned to their proper positions. (Section M1.2)

Engineering

The inspectors noted several deficiencies where actual plant trip setpoints and allowable values were inconsistent with the corresponding values contained in the Technical Safety Requirements and, in several cases, with other plant documents. In addition, the values were developed in a manner inconsistent with ISA-S67.04, "Setpoints for Nuclear Safety Related Instrumentation," to which the certificatee has committed. (Section E1.1)

Plant Support

The inspectors concluded that an emergency preparedness (EP) exercise conducted during the inspection period was realistic for the site and adequately tested the certificatee's staff on their EP training and use of procedures. The EP exercise revealed areas for improvement that were comprehensively identified by the certificatee and documented in their corrective action program for further review and resolution. The October 31 emergency response to Building X-326 revealed a strong questioning attitude by operations staff and a timely and thorough response to the event. Corrective actions were identified and implemented in a timely and thorough manner. (Section P1.1)

The inspectors' selected review and observation of shipping activities indicated that plant staff continued to implement a transportation safety program in accordance with the Department of Transportation and NRC regulations. (Section A1.1)

No issues were identified during a review of the storage of radioactive waste. (Section W1.1)

Training and associated records were generally adequate. One issue regarding course titles that did not match those specified in the Safety Analysis Report was identified, and a related issue of failing to enter the issue in the corrective actions process was identified. (Section I5.1)

Report Details

I. Operations

O1 Conduct of Operations

O1.1 Use of Operator Aids

a. Inspection Scope (88100)

The inspectors reviewed the use of operator aids by operations staff in the plant.

b. Observations and Findings

On November 2, during a routine walkdown and a subsequent followup in Building X-333, the inspectors identified several deficiencies regarding the use of operator aids (OAs) by operations staff including the following:

- at cold recovery, a copy of a graph taken from an approved in-hand procedure that was not being tracked as an OA;
- a “post-it” note next to a manometer in the Area Control Room (ACR) used to delineate the expected range of the instrument;
- a copy of an unapproved revision to a plant drawing located in the ACR that was not controlled as an OA although the previous revision was listed as being an active OA;
- OAs that had numbers assigned them that were not tracked in the log book; and
- OAs listed in the log book as active but non-existent in the plant.

During further review, the inspectors noted a weakness in the governing procedure in that it did not require that OAs be reviewed to ensure that they did not deviate from the requirements of the certificate. In addition, the procedure did not require that OAs extracted from approved drawings and procedures be updated when those documents were revised. The inspectors also identified that the annual self-assessment required by the procedure was not rigorous enough to identify the deficiencies discussed above.

As an immediate corrective action, operations performed a plant-wide review and identified additional deficiencies regarding the use of OAs. In addition, the self-assessment was revised to accurately reflect the specific requirements of the governing procedure and a lessons-learned bulletin was being developed to instruct operations staff of the procedural requirements. Plant staff was also evaluating a method for ensuring that OAs were updated when their source documents were revised.

Technical Safety Requirement (TSR) 3.9.1 required, in part, that written procedures shall be prepared, reviewed, approved, and implemented for activities described in Appendix A to Safety Analysis Report (SAR) Section 6.11. SAR Section 6.11, Appendix A, described document control as an activity for which procedures shall be implemented. In addition, Procedure XP2-US-FO1109, "COP-9 Control Of Operator Aids," described the requirements for review, approval, and tracking OAs. Contrary to the above, on November 2, the inspectors identified several examples of failure by operations staff to implement the requirements of Procedure XP2-US-FO1109 in that OAs were not properly reviewed, approved, and tracked.
This is a **Violation (70-7002/2000011-01a).**

c. Conclusions

The inspectors identified several deficiencies regarding the use of OAs by plant staff and a weakness with the required annual self-assessment in that it was not rigorous enough to identify the deficiencies. Plant staff took appropriate corrective action in response to address the issues.

O8 Miscellaneous Operations Issues

O8.1 Certificatee Event Reports (90712)

None during the inspection period.

O8.2 Bulletin 91-01 Reports (97012)

The certificatee made the following report pursuant to Bulletin 91-01 during the inspection period. The inspectors reviewed any immediate Nuclear Criticality Safety (NCS) concerns associated with the report at the time of the initial verbal notification. Any significant issues emerging from these reviews are discussed in separate sections of this report or in future inspection reports.

Number	Date	Title
37537	11/20/00	4-Hour Report - NCS violation; Nuclear Criticality Safety Approval (NCSA) was determined to be deficient in that it did not adequately consider conditions created by a 1-1/2 inch dike installed in several areas of Building X-705 and its relationship to dry active waste containers which have drainage holes in the bottom of the containers.

II. Maintenance

M1 Conduct of Maintenance Activities

M1.1 Observation of Maintenance and Surveillance Activities

a. Inspection Scope (88103)

The inspectors assessed whether general maintenance operations, surveillance tests, and calibrations were being conducted in accordance with certificate requirements and approved procedures. Work reviewed included:

- Maintenance Work Package 0022973-01, “Repair/Replace CADP Smokehead 31-2 S104”;
- Maintenance Work Package 0020637-01, “Replace 29-6-5 4B Seal”; and
- Maintenance Work Package 0027288-01, “Troubleshoot/Repair/Calibrate PI-705 Pressure Loop.”

b. Observations and Findings

The inspectors noted that the work packages reviewed were detailed and contained appropriate work instructions, procedures, and post-maintenance testing requirements. However, some deficiencies were identified while reviewing work activities:

- On November 2 while observing the completion of the work package documentation for the seal replacement, the inspectors noted that the maintenance mechanic who performed the motor shaft alignment for the involved stage was transferring data from a piece of cardboard to the designated data sheet in the work package. At the time, the worker was not at the job site. The inspectors inquired about the permissibility of copying data from scratch paper to the procedure. The worker informed the inspectors that, to the best of the worker's knowledge, the practice was acceptable and was used because the work area was potentially contaminated. The worker elaborated that normally another worker would be used as a procedure reader and data recorder, but no other worker was available for the job.

  The inspectors subsequently reviewed work control procedures and concluded that the failure to record the data in the procedure as the work progressed was not acceptable. Additionally, the inspectors verified that the procedure being used was “in-hand," which meant that the worker was required to use and fill in the procedure continuously. The inspectors contacted the maintenance supervisor and discussed the issue, and the supervisor agreed that the practice was unacceptable. Subsequently, the supervisor documented the issue in Problem Report (PR)-PTS-00-05111 and required that the work be re-done.

  As corrective action, maintenance management performed stand-down training for all personnel on the proper use of procedures. In addition, management indicated that as a result of past problems, a training package had been developed to reinforce the expectations for using procedures and recording

steps performed and data taken as part of the work. The inspectors noted that the training package addressed the scenario observed. The plant staff were in the process of providing the training to plant functional managers who were then tasked with providing the training to those reporting to them.

TSR 3.9.1 required, in part, that written procedures shall be implemented for activities described in Appendix A to SAR, Section 6.11. SAR Section 6.11, Appendix A, described procedures management as an activity for which procedures shall be implemented. In addition, Step 6.2.3 of Procedure UE2-PS-PS1034, “Use of Procedures,” required that “in-hand” procedures be taken to the work location, each step be read before performing it, and each step be performed in order unless deviation is allowed by the procedure. Contrary to the above, on November 2, staff executing “in-hand" Procedure XP3-GS-MM4070, "Shaft Alignment," did not have the procedure at the work station and, therefore, did not read each step before performing it as required by Procedure UE2-PS-PS1034. In addition, steps were not performed in order as required, as data was not recorded in the procedure but was instead recorded on scratch paper and later transferred to the procedure.
This is a **Violation (VIO 70-7002/2000011-01b).**

- On November 7 while reviewing Work Package 0027288-01 in Building X-343, the inspectors noted that a lifted lead and jumper log was not used for the replacement of pressure transducer PT-705. During discussions with maintenance personnel, the inspectors identified there was confusion regarding the use of the log because the transducer had a plug instead of traditional leads. Maintenance management confirmed that the log should have been used to install the transducer as stated in the governing procedure.

  As corrective action, management initiated PR-PTS-00-5182 to document the issue, verified that the transducer was properly installed, and completed the log as a “late entry.” Management also conducted stand-down training with applicable personnel regarding the use of lifted lead and jumper logs.

  TSR 3.9.1 required, in part, that written procedures shall be implemented for activities described in Appendix A to SAR, Section 6.11. SAR Section 6.11, Appendix A described equipment control as an activity for which procedures shall be implemented. In addition, Step 7.2 of Procedure XP2-GP-GP1033, “Lifted Leads And Jumpers,” required that leads that were lifted be recorded on the lifted lead and jumper sheet. Step 3.1 of the procedure stated that plugs were considered lifted leads. Contrary to the above, on November 7, the inspectors identified that maintenance staff did not use a lifted lead and jumper log as required by XP2-GP-GP1033 during replacement of transducer PT-705 in Building X-343. This is a **Violation (VIO 70-7002/2000011-01c)**

c. Conclusions

The work packages reviewed were detailed and contained appropriate work instructions, procedures, and post-maintenance testing requirements. However, violations for failing to use a lifted lead and jumper log and failing to follow the "in-hand" requirements during execution of a procedure were identified. Plant management took appropriate action to address the issues.

M1.2 Mispositioned Smoke Detector

a. Inspection Scope (88102)

The inspectors assessed the operability status of safety systems during routine plant walkdowns.

b. Observations and Findings

On October 19 during a routine walkdown in Building X-333, the inspectors observed that a Cascade Automatic Data Processing (CADP) smokehead was not positioned properly above the Low Assay Withdrawal compressors. The smokehead was positioned upside down on a partition between two compressors instead of hanging in its normal position on a swivel bar. The inspectors notified building management who documented the issue on PR-PTS-0004924.

During followup, plant staff determined that the smokehead had been moved to perform testing on the low-voltage smokeheads that were located in the same general area but was not properly repositioned afterwards. An engineering evaluation was prepared that determined that the smokehead would have been able to perform its intended function in the unauthorized position. In response, plant staff initiated a work request to reposition the smokehead and prepared a daily operating instruction that required the Plant Shift Superintendent (PSS) be contacted for an operability determination prior to relocating or moving a smokehead. Guidance was also to be provided to work planners to ensure that a “quality” work package was prepared prior to relocating a smokehead and that the package contained steps to ensure the smokehead was returned to its proper position. The “quality” work package would ensure that operations would perform an operability determination when reviewing for work start approval.

c. Conclusions

The inspectors identified a smokehead that was not returned to its proper position after having been moved to support a maintenance activity. In response, plant staff initiated appropriate action to ensure that an operability determination was performed prior to repositioning smokeheads in the future and that smokeheads were returned to their proper positions.

III. Engineering

E1 Conduct of Engineering

E1.1 Limiting Control Settings and Allowable Values

a. Inspection Scope (88100)

In response to issues related to the installation of an upgraded pressure transmitter on Autoclave Number 2 in Building X-344A, the inspectors reviewed the plant's interpretation and use of limiting control settings (LCSs) and allowable values contained in the Technical Safety Requirements (TSRs).

b. Observations and Findings

Limiting Control Settings

Limiting control settings (LCSs) are defined in 10 CFR 76.4 as settings for automatic alarm or protective devices related to those variables having significant safety functions. Further, 10 CFR 76.87 (d) states that TSRs must include LCSs which satisfied the following:

> Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded.

The NRC staff has historically addressed limiting safety system settings (LSSSs) (nuclear power reactor's equivalent of LCSs) by Regulatory Guide 1.105, "Instrument Setpoints for Safety-Related Systems," and its limited endorsement of ISA-S67.04, "Setpoints for Nuclear Safety Related Instrumentation." Typically, by following the guidance in the ISA standard, an LSSS would be developed by first determining the specific safety or analytical limit for the process variable (pressure, temperature, etc.) under consideration. Then, recognizing that all instrumentation had errors and/or uncertainties (calibration tolerance, test equipment accuracy, setpoint drift with time, environmental errors, etc.) associated with measuring a process variable, the errors would be identified and combined in some credible, statistical manner (algebraic summation, square root of the sum of the squares, etc.) to determine a total channel uncertainty term. This term would then be either added to the safety or analytical limit (for a protective action concerning a decreasing variable) or subtracted from the safety or analytical limit (for a protective action concerning an increasing variable). The result would be the LSSS except in cases where extra conservatism (margin) would be added by moving the LSSS away from the safety or analytical limit by an additional amount (not an error or uncertainty).

An instrument channel's LSSS (LCS) calculated as discussed above satisfied the definition of an LCS in that it accounted (was chosen) for all the instrument errors (in the setpoint methodology calculations) and in that way ensured the protective action occurred before the specific variable being monitored (or parameter associated with the variable being monitored) exceeded the safety limit.

Typically, the actual setpoint in a plant was equivalent to the LSSS (LCS) for a given process variable, since the LCS was defined to be a setting and the setpoint methodology evaluated and justified the adequacy of instrumentation uncertainties and margin associated with that setting. This was supported by Figure 1 and related discussion in the 1988 version of the ISA standard, which showed the trip setpoint as an LSSS, and Section 4.3.1 in the 1994 version which stated: “The trip setpoint should be the value that the final setpoint device is set to actuate.”

Section 1.2.12 of the plant's TSRs defined an LCS as a setting for automatic alarm or protective devices related to those variables having a significant safety function. An example of a TSR which supported an interpretation that an LCS was equivalent to the actual plant trip setpoint was noted in TSR 2.1.3.4 for the autoclave shell high steam pressure shutdown instrumentation which stated: “LCS: Steam pressure set points shall

be in accordance with table below." Also, related Surveillance Requirement (SR) 2.1.3.4.1 stated: "Perform channel functional test to verify the autoclave steam supply valves will close when the pressure exceeds the LCS set point" and SR 2.1.3.4.2 stated: "Calibrate shell high pressure steam shutdown instrumentation at the installed LCS set point." Also, each time the setpoint was changed, SR 2.1 3.4.3 required: "Calibrate shell high pressure steam shutdown instrumentation at new LCS set point."

Although the certificatee was committed to the 1988 version of the ISA standard and referred to that version and the 1994 version in correspondence to the NRC, the certificatee had indicated during discussions with the inspectors that a lower setpoint, not equal to the LCS, could be used in the plant without changing the related TSR. The inspectors' review determined that these statements were supported by notes found in plant documents such as SPT-344-04, Revision 1, "Setpoint Analysis and Calculation for X-344 Upgrade Autoclaves, Autoclave Shell High Steam Pressure Shutdown," which stated: "Calculations apply to the LCO setpoint. An operating setpoint may be set at a lower level..." Also the same note in Revision 0 of SPT-344-04 stated: "Calculations apply to the setpoint only. Calibrations at points other than the setpoint should be based on past experience or other special consideration as necessary."

Although the inspectors found that the plant's actual trip setpoint for the autoclave shell high steam pressure shutdown instrumentation agreed with the LCS value specified in TSR 2.1.3.4, the situation did exist where the certificatee could take the actions allowed by the above notes resulting in the plant not meeting SR 2.1.3.4.2 and SR 2.1.3.4.3. Also, the TSR LCS would no longer be controlling plant safety if the licensee did change a setpoint in the plant without changing the TSR LCS. The plant staff's further review of the intent, supported by notes in the two versions of SPT-344-04, to use actual trip setpoints that did not coincide with the specified TSR LCS and that did not follow the commitment to ISA-S67.04 is an **Inspection Follow-up Item (IFI 70-7002/2000011-02).**

TSR 2.1.3.5 encompassed the requirements for the trip setpoint for autoclave shell high pressure containment shutdown. Specifically, SR 2.1.3.5.1 required the certificatee to: "Perform channel functional test to verify the High Pressure Containment system will actuate at or below 15 psig." Also, SR 2.1.3.5.2 stated: "Calibrate shell high pressure shutdown instrumentation at ≤ 15 psig." Although the basis for TSR 2.1.3.5 implied that 15 psig was the setpoint, the "≤ " allowed somewhat of an engineering debate. Typically, this would have been interpreted to be a setpoint of 15 psig with a calibration error permitted on the low side. In other words, the channel would be set as close to 15 psig as possible with slight deviation permitted on the low side. The inspector's review indicated that the plant's calibration procedure XP4-TE-IM6703, Revision 4, "X-344 Autoclave 1 Pressure Calibration," specified a setpoint of 8 psig for this variable which was substantially less than 15 psig.

The shell high pressure containment shutdown instrumentation had an analytical limit of 165 psig with ≤ 15 psig specified as the setpoint in the TSR; an actual setpoint of 8 psig specified in plant procedures; a total channel uncertainty of 0.2 psig calculated in SPT-344-11, "Setpoint Analysis and Calculation for X-344 Upgrade Autoclaves, Autoclave Shell High Pressure Containment Shutdown"; and a resulting margin of 156.8 psi which appeared to the inspectors to be very excessive. In addition, according to page 3.2-8 of the plant's Safety Analysis Report (SAR), the steam pressure in the supply line was controlled to a maximum of 8 psig or 2.5 psig (depending on the category of cylinder) for low steam flow and 12 psig for high flow. Since these values

could overlap the setpoint, spurious actuations of the protective systems were possible resulting in unnecessary challenges to these safety systems. According to the 1994 version of the ISA standard, use of excessive margin was to be avoided, and the ISA stated:

> It is prudent to evaluate setpoint calculations to assure they are not overly conservative. Overly conservative setpoints can be restrictive to plant operation or may reduce safety by unnecessarily increasing the frequency of safety system actuation.

Discussions with plant staff indicated that the incidence of spurious actuations of the safety systems associated with the autoclave shell pressure instrumentation was low, and the inspectors did not consider spurious actuation to be a safety concern. However, the plant's use of excessive margin in the selection of trip setpoints was an observation by the inspector discussed with the certificatee.

During the inspectors' review of XP4-TE-UH6717A, "X-344 Autoclave NSU Quarterly Tests," it was noted that an incorrect tolerance was used. Also, in XP4-TE-IM6703, "X-344 Autoclave 1 Pressure Calibration," additional discrepancies were noted for as-found tolerances. No immediate operability issues were identified as a result of the discrepancies. In response, the certificatee issued two problem reports (PRs) recommending that maintenance and operations calibration and surveillance procedures for autoclaves be reviewed and updated (if needed) to ensure the correct flowdown of setpoints and tolerances for Quality and other TSR systems. Pending completion and inspector review of the recommended action contained in the PRs, this is an **Inspection Follow-up Item (IFI 70-7002/2000011-03)**.

Allowable Values

The NRC staff had historically addressed the term allowable value by Regulatory Guide 1.105's limited endorsement of ISA-S67.04. The definition of an allowable value in the 1988 version of the ISA standard was: "A limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is declared inoperable and corrective action must be taken."

Typically, an LSSS/LCS would be developed by following the guidance in the ISA standard as discussed above. The allowable value would be calculated by first identifying the instrumentation uncertainties specifically associated with periodic testing such as test equipment accuracy, calibration tolerance, and drift. This would be combined in some credible, statistical manner to determine a channel uncertainty term associated with the specific periodic test used to check channel performance. This term would then be either added to the LSSS (for a protective action concerning an increasing variable) or subtracted from the LSSS (for a protective action concerning a decreasing variable). The result would be the allowable value and would be used for judging the channel's past performance and operability. Specifically, if the actual value was measured at which the channel tripped (as-found setpoint), it could be determined if the channel's performance (such as drift) during the past surveillance interval was acceptable by comparing that value to the specified allowable value.

Section 1.2.3 of the plant's TSR used the ISA standard's definition for an allowable value. A specific TSR such as 2.1.3.4 stated in the basis that:

> The LCS set points of 8 psig and 2.5 psig as they correspond to cylinder categories were established such that when instrument drift and other calibration uncertainties (2 psig) associated with the installed instrumentation is added to the LCS setpoint (10 psig = 240 ^{o}F and 4.5 psig = 226 ^{o}F) there is an adequate safety margin. The 10 psig and 4.5 psig values correspond to the "Allowable Value" as defined in ANSI/ISA-S67.04-1988.

The inspectors' review of SPT 344-04, Revision 0, the setpoint analysis and calculation for the old (prior to upgrade) autoclave shell high steam pressure shutdown instrumentation for X-344 Autoclaves 1 thru 4, determined that the total channel uncertainty term (also equal to the channel uncertainty term associated with periodic testing) was 0.7 psi which provided a 1.3 psi margin (not an instrument uncertainty) in the TSR allowable value (2.0 - 0.7 = 1.3). The inspector's review of the quarterly surveillance procedure (XP4-TE-UH6717, Revision 4) indicated that 0.7 psig (not 2.0 psig as indicted in the TSR) was the acceptance value used to determine operability for this instrumentation.

Although the certificatee was committed to the 1988 version of the ISA standard, the plant's operability determination, based on actual equipment uncertainties, was much tighter than the TSR's operability which was based on an allowable value that contained margin that was unrelated to actual equipment performance. Once again, the plant procedures and not the TSR were controlling plant safety and safety-related equipment performance. Also, the basis for TSR 2.1.3.4 was not correct since 2 psig was identified as only instrument drift and other calibration uncertainties and not margin. In response to the inspectors' questions, the licensee stated that it would consider revising the TSR basis to more closely follow its practice of including margin in the allowable value.

As another example of this situation, the basis for TSR 2.1.3.5 stated that there was a 2 psi tolerance to comply with the definition of allowable value in the ISA standard. For the specified setpoint (discussed above) of 15 psig, this resulted in an allowable value (AV) of 17 psig. The uncertainty term for periodic testing of the old instrumentation was again 0.7 psi, leading to a margin of 1.3 psi as above. When the plant used the lower setpoint (8.0 psig instead of 15 psig as discussed above), the AV was not changed. Thus the margin in the AV then became 8.3 psig which put the AV even further away from a realistic, actual instrumentation performance measure (0.7 psi) used for the plant operability determination. For the upgraded instrumentation, the value increased to 8.8 psi margin.

Section 4.3.2 of the 1988 version of the ISA standard stated:

> The uncertainties of that portion of the instrument channel being tested to be used to determine the allowance between the trip setpoint and the Allowable Values are:
>
> 1. Instrument Calibration Uncertainties
> 2. Instrument Uncertainties During Normal Operation
> 3. Instrument drift.

This is further amplified by Part II to the 1994 version of the ISA standard which stated: "The allowable value also provides a means to identify unacceptable instrument performance that may require corrective action." Further, it stated: "The allowance

between the allowable value and the trip setpoint should contain that portion of the instrument channel being tested for the surveillance interval...and should account for no more than a) drift (based on surveillance interval); b) instrument calibration uncertainties for the portion of the instrument channel tested; and c) instrument uncertainties during normal operation that are measured during testing..."

It also stated: "Recall that the AV is a parameter to be used by the plant to verify channel performance at prescribed surveillance intervals."

The actual plant setpoints and AVs for the plant's instrumentation were controlled by setpoint change control procedure XP2-EG-CF1075, Revision 0. The first paragraph of Appendix B of that document stated: "The methods used to calculate setpoints and allowable values set forth in this appendix complies with the intent of ISA-S67.04 Part I and ISA-S67.04 Part 2 unless otherwise noted." Specifically, a note on page 6 of Appendix B stated: "This is in agreement with Method 3 of ISA RP-67.04 Part II in establishing an allowable value." Since the inspectors' review of Method 3 in Part II of that recommended practice indicated that margin was not included in the determination of the AV and the licensee's methodology did include margin as shown on page 7 of Appendix B, the plant's TSR AV determination methodology and specific AVs determined by the use of that methodology were not in agreement with the ISA standard. The situation was further compounded by other confusing statements such as found on page 6 of XP2-EG-CF1075 which stated:

> The Allowable Value is a limiting value that the trip setpoint may have when tested periodically, beyond which appropriate action shall be taken. This is normally the setpoint plus or minus the allowable test tolerance.

Since there was no mention of margin in the above statement, the inspectors questioned the certificatee about this statement. In response, the certificatee stated that this AV was not the TSR AV but the AV used by the plant for calibration purposes. Also, on that same page of the procedure, the region between the AV and the trip setpoint was defined to be "the region where the equipment is operable...If this region is exceeded, it may mean the instrument is not or has not performed within the assumption of the setpoint calculation and should be evaluated for operability." This statement was related to a figure which was intended to show the basic relationships between a setpoint and various limits as defined in ISA-S67.04. In other words, the figure and the statements on page 6 of XP2-EG-CF1075 agreed with the ISA standard and the TSR basis statements referring to the ISA standard, but the certificatee's verbal response did not.

Also page 4 of Appendix A to XP2-EG-CF1075 stated:

> TSR Allowable Value - The setpoint value as defined in the TSR. If this TSR value plus (+) or minus (-) any assigned as found tolerance is exceeded, a plant change request should determine the need for a safety evaluation.

The inspectors questioned the meaning of this statement and in response, the certificatee stated that this section needed to be rewritten for clarity. Plant staff's resolution regarding identification and use of tighter operability determinations coupled

with the failure to update the corresponding AVs in the TSRs to reflect true instrumentation performance parameters and the commitment to ISA-S67.04 is an **Inspection Follow-up Item (IFI 70-7002/2000011-04).**

c. Conclusions

The inspector noted several deficiencies where actual plant trip setpoints and AVs were inconsistent with the corresponding values contained in the plant's TSRs and, in several cases, with other plant documents. In addition, the values were developed in a manner inconsistent with ISA-S67.04 to which the certificatee has committed.

IV. Plant Support

P1 Emergency Preparedness Program

P1.1 Emergency Preparedness Observations

a. Inspection Scope (88050)

The inspectors reviewed the certificatee's emergency preparedness (EP) program exercise to assess plant staff's state of operational readiness. The inspectors also observed the response of EP personnel to an event at Building X-326.

b. Observations and Findings

Emergency Exercise

On October 18 an EP drill was conducted that simulated an inadvertent criticality in Building X-705. The scenario involved uranium-bearing solution that accumulated in an unsafe geometry, a wheelbarrow located in the recovery area. The criticality resulted in radiation exposure and injuries to some workers located in the building at the time of the simulated accident. Following the prompt criticality, the radiation hazard remained in the area which was designed to complicate the response. The inspectors determined that the scenario was realistic and adequately challenging for the EP organization. The inspectors did not observe any pre-staging of personnel or equipment prior to the EP drill.

During the EP drill, the inspectors observed the activities of the responders, including plant supervision, the station fire department, medical, and security staff. The inspectors noted an "Alert" was declared as required by procedure. The inspectors identified some areas for improvement in the certificatee's response. A "safe route" out of the facility for evacuating personnel was not clearly defined. In addition, the command post was relocated twice to ensure that it was located in a low dose area; however, relocation added confusion and some delay to the response. The inspectors also observed activities in the Emergency Operations Center (EOC). The certificatee performed a debrief immediately following the event and an exercise critique the following day. The inspectors concluded that the certificatee's self assessment addressed the problems observed by the inspector. The certificatee performed a comprehensive assessment of the exercise and identified areas for improvement. The problems identified were entered into the corrective action program. Examples of problems identified were: inadequate communication between the Crisis Manager and

the Incident Commander; adequacy of the technical depth of the EOC cadres; the appropriate application of TSR 1.6.4; and safe access routes to the EOC.

Emergency Response

On October 31 the inspectors also observed an emergency response to Building X-326. Smoke was reported coming from side purge cell 25-7-4. Firefighters responded immediately and the EOC was activated for technical support. The firefighters located the source of the smoke and diagnosed the problem as an electrical duct heater failure. Operators isolated the oil supply and the electrical power to the heaters.

The inspectors observed the emergency response, communications, and use of procedures. At the EOC, the inspectors observed the conclusion of the response and the debrief. The inspectors observed difficulties in the lines of communication between the firefighters and the fire commander, both using radios. The firefighters, who were wearing self-contained breathing apparatus (SCBA), were very difficult to understand when speaking and did not use standardized language for questions, commands, or responses. This observation and others were captured in the certificatee's corrective action program.

c. Conclusions

The inspectors concluded that the October 18 EP exercise was realistic for the site and adequately tested the certificatee's staff on their EP training and use of procedures. The exercise revealed areas for improvement that were comprehensively identified by the certificatee and documented in their corrective action program for further review and resolution. The October 31 emergency response revealed a strong questioning attitude by operations staff and a timely and thorough response to the event. Corrective actions were identified and implemented in a timely and thorough manner.

A1 Conduct of Transportation Activities

A1.1 Transportation of Radioactive Materials

a. Inspection Scope (86740)

The inspectors reviewed shipping papers, selected surveys, and preparation for shipment of radioactive materials, including 2.5-ton and 10-ton cylinders and low-level radioactive waste in B-25 containers.

b. Observations and Findings

The inspectors noted that the shipments of radioactive materials were properly braced and the transport vehicles properly placarded. Packages observed, such as the cylinder overpacks for fissile uranium hexafluoride (UF_6) shipments, had the appropriate labels and markings on the exterior. The transport indices also conformed to the requirements of the Department of Transportation (DOT) and 10 CFR 71. Plant staff preparing the shipments were knowledgeable of the transportation requirements. In addition, the inspectors noted that the traffic department did a final walk-around of the vehicles and packages before the truck drivers were cleared to leave the site.

The survey records reviewed indicated that the appropriate radiation and contamination limits in the DOT regulations were met for exclusive-use shipments. The dose rates on contact and at one meter for the overpacks were less than the 200-millirem-per-hour and 10-millirem-per-hour limits. Contamination smear results were all less than detectable for both alpha and beta-gamma contamination. The surveys of the truck cabs and trailers did not identify any contamination or elevated radiation levels in those areas.

c. Conclusions

The inspectors' selected reviews and observations of shipping activities indicated that plant staff continued to implement a transportation safety program in accordance with DOT and NRC regulations.

W1 Radioactive Waste Management

W1.1 Review of Radioactive Waste Program

a. Inspection Scope (88035)

The inspectors reviewed the storage of radioactive waste. The review included assessment of the criticality controls, the material condition of the facility, and discussions with cognizant personnel.

b. Observations and Findings

The material condition of the waste storage areas was good. No evidence of water intrusion into the building or significant degradation of equipment or containers was noted. The personnel in the area were knowledgeable. The inspectors verified compliance with posted criticality controls by randomly checking that uranium mass and assay were quantified and documented on tags attached to the containers. The inspectors reviewed ten randomly selected records for waste storage containers and verified that contents were traceable to the container. The physical and chemical characteristics of the container were properly documented.

c. Conclusions

No issues were identified during a review of the storage of radioactive waste.

I5.1 Training

I5.1 Review of Training Program

a. Inspection Scope (88010)

The inspectors reviewed a sample of required employee training to verify compliance with regulatory requirements.

b. Observations and Findings

The inspectors discussed the training program with the Training Manager and reviewed a sample of required training course materials. In discussion with the inspectors, the Training Manager explained that all required training was developed using the systems approach to training. The inspectors noted that the training course material included descriptive learning objectives, the effective use of learning tools, and a final test that demonstrated the student's knowledge of the learning objectives.

The inspectors noted training course titles were not descriptive. Specifically, the inspectors were required to review the training courses' objectives to ensure that the training courses required by SAR Section 6.6, "Training," were completed because of some vague training course titles. The Training Manager explained that a self-assessment had independently identified the title issue. However, the inspectors noted that the training staff had not issued a problem report (PR) on the issue until questioned by the inspectors. Subsequently, plant staff prepared PR-PRTS-00-05098 to document the issue. In addition, the Training Manager prepared a lessons-learned report to heighten awareness of the need to submit PRs associated with findings and recommendations made during self assessments. The failure to submit PRs for findings from the self-assessments constitutes a violation of minor significance.

c. Conclusions

The inspectors identified a minor training program weakness that had previously been identified during a plant staff self-assessment, but the issue was not entered into the formal corrective actions program. The inspectors concluded that the training course materials reviewed were developed in accordance with regulatory and procedural requirements.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the facility management on November 20, 2000. The facility staff acknowledged the findings presented and indicated concurrence with the facts, as stated. The inspectors asked the plant staff whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

United States Enrichment Corporation

*P. Musser, General Manager
*J. Anzelmo, Work Control Manager
D. Couser, Training & Procedures Manager
*J. Cox, Plant Services Manager
*S. Fout, Enrichment Plant Manager
*R. Helme, Engineering Manager
*R. Lawton, Nuclear Safety & Quality Manager
*P. Miner, Regulatory Affairs/Commitment Management Manager
R. Smith, Plant Support Manager
*M. Wayland, Maintenance Manager

*Denotes those present at the exit meeting on November 20, 2000.

INSPECTION PROCEDURES USED

IP 88100:	Plant Operations
IP 88102:	Surveillance
IP 88103:	Maintenance
IP 88010:	Training
IP 88035:	Radioactive Waste Management
IP 88050:	Emergency Preparedness
IP 86740:	Transportation of Radioactive Materials
IP 88100:	Engineering
IP 90712	In-office reviews of Written Reports on Non-routine Events

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened	Item Type	Summary
70-7002/2000011-02	IFI	Plant staff's further review of the intent to use actual trip setpoints that did not coincide with the specified TSR LCS and that did not follow the commitment to ISA-S67.04.
70-7002/2000011-03	IFI	Plant staff's review of maintenance and operations calibration and surveillance procedures for autoclaves to ensure the correct flowdown of setpoints and tolerances for Quality and other TSR systems.
70-7002/2000011-04	IFI	Plant staff's resolution regarding identification and use of tighter operability determinations coupled with the failure to update the corresponding AVs in the TSRs to reflect true instrumentation performance parameters and the commitment to ISA-S67.04.

Closed

70-7002/2000011-01 VIO Failure to follow plant procedures.

Discussed

None

LIST OF ACRONYMS USED

ACR	Area Control Room
ADAMS	Agencywide Documents Access and Management System
CAAS	Criticality Accident Alarm System
CADP	Cascade Automatic Data Processing
CER	Certificate Event Report
CFR	Code of Federal Regulations
DNMS	Division of Nuclear Material Safety
DOE	Department of Energy
DOT	Department of Transportation
EOC	Emergency Operations Center
EP	Emergency Preparedness
IFI	Inspection Follow-up Item
LAW	Low Assay Withdrawal
LCS	Limiting Control Setting
LSSS	Limiting Safety System Setting
NCS	Nuclear Criticality Safety
NCSA	Nuclear Criticality Safety Approval
NRC	Nuclear Regulatory Commission
OA	Operator Aid
PARS	Publicly Available Records
PERR	Public Electronic Reading Room
PR	Problem Report
PSS	Plant Shift Superintendent
PT	Pressure Transducer
SAR	Safety Analysis Report
SCBA	Self-contained Breathing Apparatus
SR	Surveillance Requirement
TSR	Technical Safety Requirements
UF_6	Uranium Hexafluoride
URI	Unresolved Item
USEC	United States Enrichment Corporation
VIO	Violation