NUREG-1450

# Potential Criticality Accident at the General Electric Nuclear Fuel and Component Manufacturing Facility, May 29, 1991

U.S. Nuclear Regulatory Commission

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## ABSTRACT

At the General Electric Nuclear Fuel and Component Manufacturing facility, located near Wilmington, North Carolina, on May 28 and 29, 1991, approximately 150 kilograms of uranium were inadvertently transferred from safe process tanks to an unsafe tank located at the waste treatment facility, thus creating the potential for a localized criticality safety problem. The excess uranium was ultimately safely recovered when the tank contents were centrifuged to remove the uranium-bearing material. Subsequently, the U.S. Nuclear Regulatory Commission dispatched an Incident Investigation Team to determine what happened, to identify probable causes, and to make appropriate findings and conclusions. This report describes the incident, the methodology used by the team in its investigation, and presents the team's findings and conclusions.

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# ACRONYMS

| | |
|---|---|
| AA | assigned airborne |
| ABC | ammonium bicarbonate |
| ADU | ammonium diuranate |
| AEC | active engineered control |
| AF | aqueous feed |
| ASMU | aqueous solution make-up unit |
| ASTM | American Society for Testing and Materials |
| AW | aqueous waste |
| AWQT | aqueous waste quarantine tanks |
| $CaF_2$ | calcium fluoride |
| C&IS | control and instrumentation services |
| CM | corrective maintenance |
| CRO | control room operator |
| CRT | cathode ray tube |
| CV | control valve |
| EAL | emergency action level |
| ECC | emergency control center |
| ED | emergency director |
| EDO | Office of the Executive Director for Operations |
| ENS | emergency notification system |
| EO | emergency officer |
| FCO | fuel component operation |
| FCR | facility change request |
| FCV | flow control valve |
| FMO | fuel manufacturing operation |
| g U/l | grams of uranium per liter |
| gal | gallon |
| GE | General Electric |
| gph | gallons per hour |
| gpm | gallons per minute |
| HEC | head-end concentrator |
| HOO | Headquarters Operations Officer |
| HVAC | heating, ventilation, and air conditioning |

| | |
|---|---|
| I&C | instrumentation and control |
| ICAP | inductively coupled argon plasma |
| ICP | inductively coupled plasma |
| IIT | Incident Investigation Team |
| IN | information notice |
| I/P | current to pneumatic |
| IRC | Incident Response Center |
| | |
| kg | kilogram |
| | |
| LCV | level control valve |
| LE | level element |
| LIC | level-indicating controller |
| LIT | level-indicating transmitter |
| | |
| $\mu$ | micro |
| $\mu$Ci/cc | microcuries per cubic centimeter |
| ma | milliamps |
| MDA | minimum detectable amount |
| mg | milligram |
| MIPVAX | Maintenance Improvement Program Data Base |
| MPC | maximum permissible concentration |
| mr | millirem |
| mrem | millirem (1,000th of a rem) |
| MWO | maintenance work order |
| | |
| NFCM | Nuclear Fuel and Components Manufacturing |
| $NH_4OH$ | ammonium hydroxide |
| NMSS | Office for Nuclear Material Safety and Safeguards |
| NRC | U.S. Nuclear Regulatory Commission |
| NSE | nuclear safety engineering organization |
| NSR/R | nuclear safety release/requirements |
| | |
| PM | preventive maintenance |
| P/P | practices and procedures |
| ppm | parts per million |
| | |
| QT | quarantine tank |
| | |
| RCEP | Radiological and Emergency Plan |
| REM | roentgen equivalent man |
| | |
| SAC | Solid Angle Code |
| SCO | service component operation |
| SX | solvent extraction |

| | |
|---|---|
| TK | tank |
| TLD | thermoluminescent dosimeter |
| U | uranium |
| $UF_6$ | uranium hexafluoride |
| UNH | uranyl nitrate |
| $UO_2$ | uranium dioxide |
| UPMP | Uranium Process Management Project |
| URLS | uranium recycle lagoon system |
| URU | uranium recycle unit |
| V | volts |
| VAC | ac voltage |
| VDC | dc voltage |
| VF | void fraction |
| WO | work order |
| WTF | Waste Treatment Facility |

## THE TEAM MEMBERS

Members of the NRC Incident Investigation Team for the GE-Wilmington Nuclear Criticality Safety Incident on May 29, 1991, are as follows:

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**ACKNOWLEDGEMENTS**

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# 1 EXECUTIVE SUMMARY

On May 29, 1991, at the General Electric (GE) Company's Nuclear Fuel and Components Manufacturing (NFCM) facility approximately 6 miles north of Wilmington, North Carolina, an estimated 150 kilograms (320 pounds) of uranium were inadvertently transferred to an unfavorable geometry waste treatment tank. ("Unfavorable geometry" refers to a container or vessel that can hold enough uranium to produce a criticality.) Because of the tank configuration and type and quantity of material available, there was the potential for a nuclear criticality accident. Such an accident would yield a burst of neutron and gamma radiation that would likely be fatal to anyone within 10 feet of the burst and cause radiation exposures of approximately 5 rads at 45 feet. However, there would be no expected off-site radiological impacts.

The NFCM facility is licensed by the U.S. Nuclear Regulatory Commission (NRC) to possess and use special nuclear material. An aerial photo of the facility is shown in Figure 1.1. Authorized activities include uranium hexafluoride ($UF_6$) conversion, fuel manufacturing, scrap recovery, process technology operations, laboratory operations, and waste treatment and disposal. As part of the fuel manufacturing process, GE's NFCM (the licensee) has established a Uranium Recycle Unit to recover uranium from certain waste and scrap materials. In this process, scrap materials are dissolved in nitric acid, passed through a filter, and fed to a solvent-extraction system. The recovered uranium is then returned to the fuel manufacturing process. A brief summary of the problem and sequence of events is presented below. A time line depicting the transfers of materials and related events is shown in Figure 1.2.

On the evening of May 28, 1991, the Uranium Recycle Unit control room operator noted that the interface level between the organic and aqueous phases within the solvent-extraction process could not be maintained. Although the operators became aware of the interface problem around 9:30 p.m., the problem actually started about an hour earlier but was apparently unnoticed by the operators. The interface problem was caused by a malfunction of the solvent-extraction Column A level control valve, LCV-300. When efforts by the control room operator to correct the level control valve problem were unsuccessful, attempts were made to control the process by throttling a manual valve located upstream of LCV-300. Manual throttling continued until shift turnover at 11:00 p.m.

The relief control room operator requested that maintenance investigate the problem with the level control valve. Until maintenance personnel arrived approximately two hours later, the floor operator continued to throttle the upstream valve manually. After approximately an hour and a half of troubleshooting activities, maintenance personnel concluded that the valve could not be repaired because replacement parts were unavailable. At the direction of the control room operator, maintenance personnel forced

LCV-300 open by redirecting air pressure in the valve actuator. Forcing the valve open caused the solvent-extraction process to be ineffective and had the effect of creating an open pathway for high concentrations of uranium to be transferred directly to the aqueous waste quarantine tanks.

From the onset of the problem, feed material (i.e., crude uranyl nitrate) continued to be sent to the solvent-extraction process. Aqueous waste from the solvent-extraction process was fed to two favorable geometry quarantine tanks. During a nine-hour period on May 28, 1991 and May 29, 1991, the contents of approximately nine quarantine tanks were transferred to an unfavorable geometry waste accumulation tank located outside the fuel manufacturing building. Of these nine transfers, four were made without a measurement of their uranium concentration. Transfers that were made after sampling and measurement, which showed concentrations of less than the 150 parts per million (ppm) transfer limit, were questionable because of sampling system problems. These problems were later confirmed by a calculational method (i.e., system mass balance), which showed that some of the analyzed tanks transferred had to contain uranium concentrations greater than 12,000 ppm.

At approximately 5:20 a.m. on May 29, 1991, a measured sample from the quarantine tank indicated a uranium concentration of 6977 ppm compared to the transfer limit of 150 ppm. Based on this information, the control room operator transferred the contents to a safe-geometry rework tank and then shut down the solvent-extraction process. Unaware of the uranium concentration problems, a Waste Treatment Facility operator approximately 10 minutes later pumped the contents of the 20,000-gallon waste accumulation tank to a comparable treatment tank with unfavorable geometry at the Waste Treatment Facility located approximately one quarter mile from the fuel manufacturing building. Sample results for the material in the waste treatment tank at that facility revealed a uranium concentration of 2333 ppm.

The licensee recognized the nuclear criticality potential of the problem but initially did not consider it to be an emergency condition. As a result of these high concentrations, the licensee assigned a technical evaluation team to develop nuclear criticality mitigation and uranium recovery plans. To minimize the nuclear criticality potential, operators continued air sparging (i.e., mixing) tank contents to prevent an accumulation of material in the bottom of the tank caused by precipitate settling.

The licensee advised the NRC of the incident on May 29, 1991; NRC formed and dispatched a response team to the site. Because of the nuclear criticality safety significance of the incident, the NRC upgraded the agency's response mode from Normal to Standby and activated both headquarter's and regional incident response centers. By the evening of May 29, 1991, the licensee had finalized uranium recovery plans and began to remove uranium from the tank with a centrifuge. On May 29, 1991, an open teleconferencing link was maintained among NRC headquarter's and regional incident response centers and the licensee's site. Initially, the licensee did not appear to treat the incident with the same degree of concern as the NRC. The NRC urged the licensee to consider emergency staffing levels, mitigating actions, and contingency measures. Based on this dialogue, the licensee increased staffing levels and developed various contingency measures. Although these actions were taken, the licensee continued to maintain that the incident did not meet the threshold required for implementation of their emergency plan. After continued prompting by the NRC, the licensee declared an Alert emergency classification, implemented provisions of the emergency plan, and notified Federal, State, and local offsite authorities about 6:40 a.m, on May 30, 1991.

Uranium recovery operations continued through the early morning of June 1, 1991. At this time, the contents of the waste storage tank were reduced to less than a critical mass after a portion of the tank contents were transferred to two adjacent storage tanks. The licensee terminated the alert classification at 3:20 a.m., June 1. Centrifuge operations continued until June 3, 1991, and upon completion of the operation, the licensee recovered approximately 150 kilograms of uranium.

On May 31, 1991, the NRC's Executive Director for Operations established an eight-member Incident Investigation Team (IIT), directing them to (1) fact find as to what happened (2) identify probable causes, and (3) make appropriate findings and conclusions. This report documents the team's efforts. The IIT arrived in Wilmington, North Carolina, on June 2, 1991. The team was selected based on its broad knowledge of facility event analysis, with individual members having specific knowledge of fuel fabrication operations, chemical operations, instrumentation and controls, maintenance, human factors, radiological emergency preparedness, and nuclear criticality safety.

Section 2 of this report describes the methods used by the team to collect and evaluate information about the incident taken from a variety of written records, physical examinations, demonstrations, tests, and interviews.

Section 3 provides a narrative and detailed sequence of events reconstructed from the team's analysis of data, interviews, procedural and record reviews, and system walkdowns.

Section 4 describes the facility and discusses general aspects of the fuel manufacturing operation. Specific emphasis is placed on operation of the Uranium Recycle Unit.

Section 5 provides information about the licensee's emergency preparedness plans and implementation, including offsite coordination and response and uranium recovery actions.

Section 6 discusses human factors considerations and evaluates personnel and operator-interface system performance.

Section 7 presents an evaluation of the licensee's nuclear criticality safety program and nuclear criticality safety implications of the incident.

Section 8 evaluates the instrumentation and control system.

Section 9 discusses the development and implementation of the licensee's maintenance program with particular attention given to the effect the maintenance process had on the incident.

Section 10 examines the licensee's capabilities to collect representative samples, discusses testing of the quarantine tanks sampling system, evaluates the Uranium Recycle Unit laboratory capabilities to adequately analyze samples, and discusses the mass balance for the solvent-extraction system.

Section 11 contains a discussion of related industry events and licensee incident precursors.

Section 12 discusses applicable regulatory requirements and implications of the incident on the fuel cycle regulatory process.

Section 13 presents the team findings and conclusions relative to the incident.

The IIT concludes that there are three interrelated root causes which contributed to the incident.

- There was a pervasive licensee attitude that a nuclear criticality was not a credible accident scenario. While the licensee understood and recognized that a nuclear criticality with low-enriched uranium was technically possible, and that there were regulatory requirements to establish measures to guard against such an accident, the licensee's perception was that the risk was so low that a criticality accident inherently would not happen.

- Licensee management did not provide effective guidance and oversight of licensed activities to assure that operations were conducted in a safe manner.

- There was a deep-seated production-minded orientation within the licensee organization that was not sufficiently tempered by a "safety first" attitude, particularly regarding nuclear criticality safety.

As illustrated in the following sections of the report, these basic causes manifested themselves in contributing causes, such as design deficiencies, procedural non-compliance, inadequate incident investigations, and a general deterioration of criticality controls.

The team also concluded that NRC regulatory oversight of the fuel facility was deficient in some respects. The team noted shortcomings with respect to the NRC's regulations and regulatory guidance, license and licensing process, and inspection program. This lack of sufficient oversight had the effect of contributing to a situation where safety margins eroded to the extent that the licensee had little or no latitude to accommodate operator errors or system upsets.

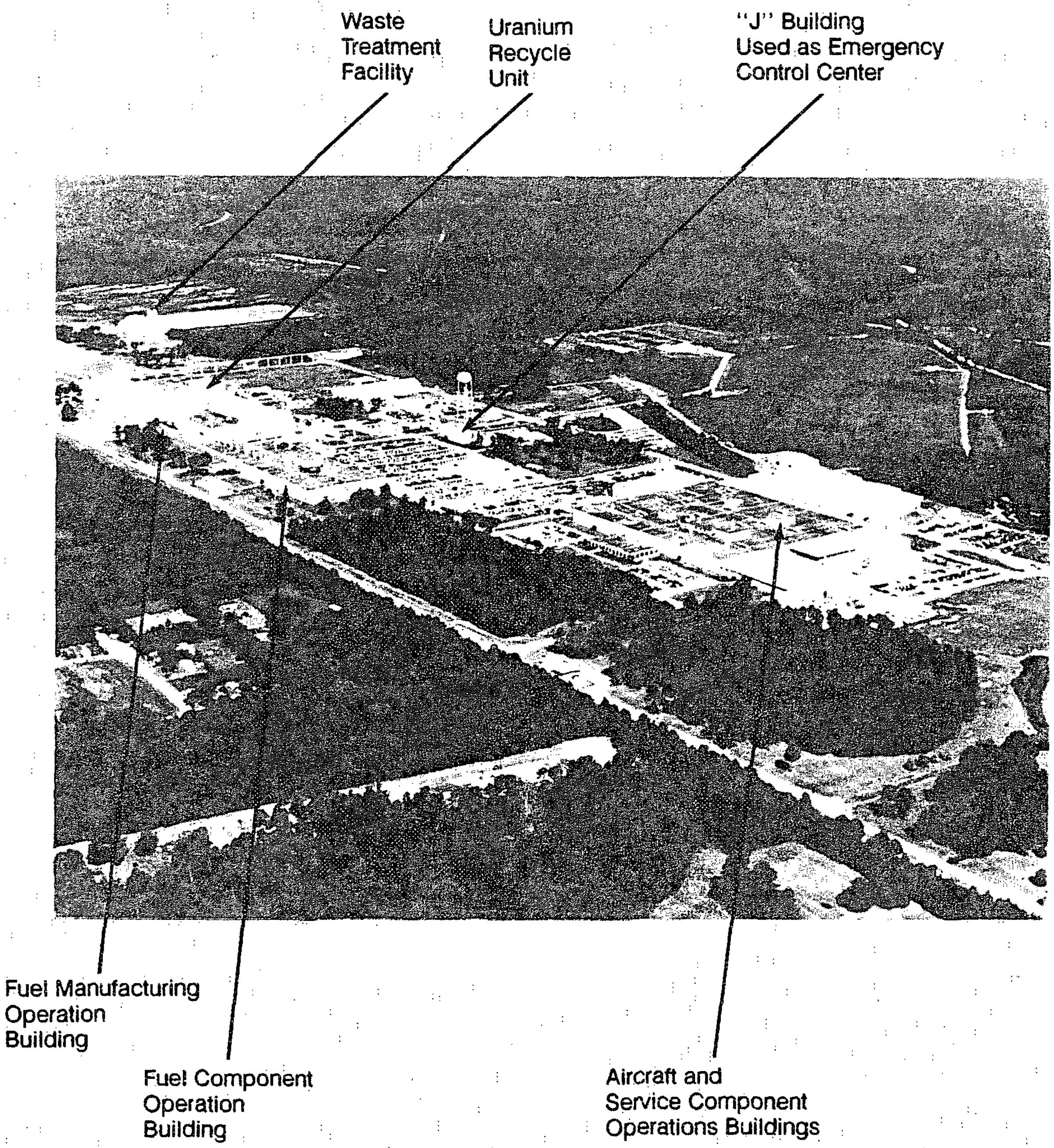


Figure 1.1 Aerial view of the General Electric – Wilmington, North Carolina, site

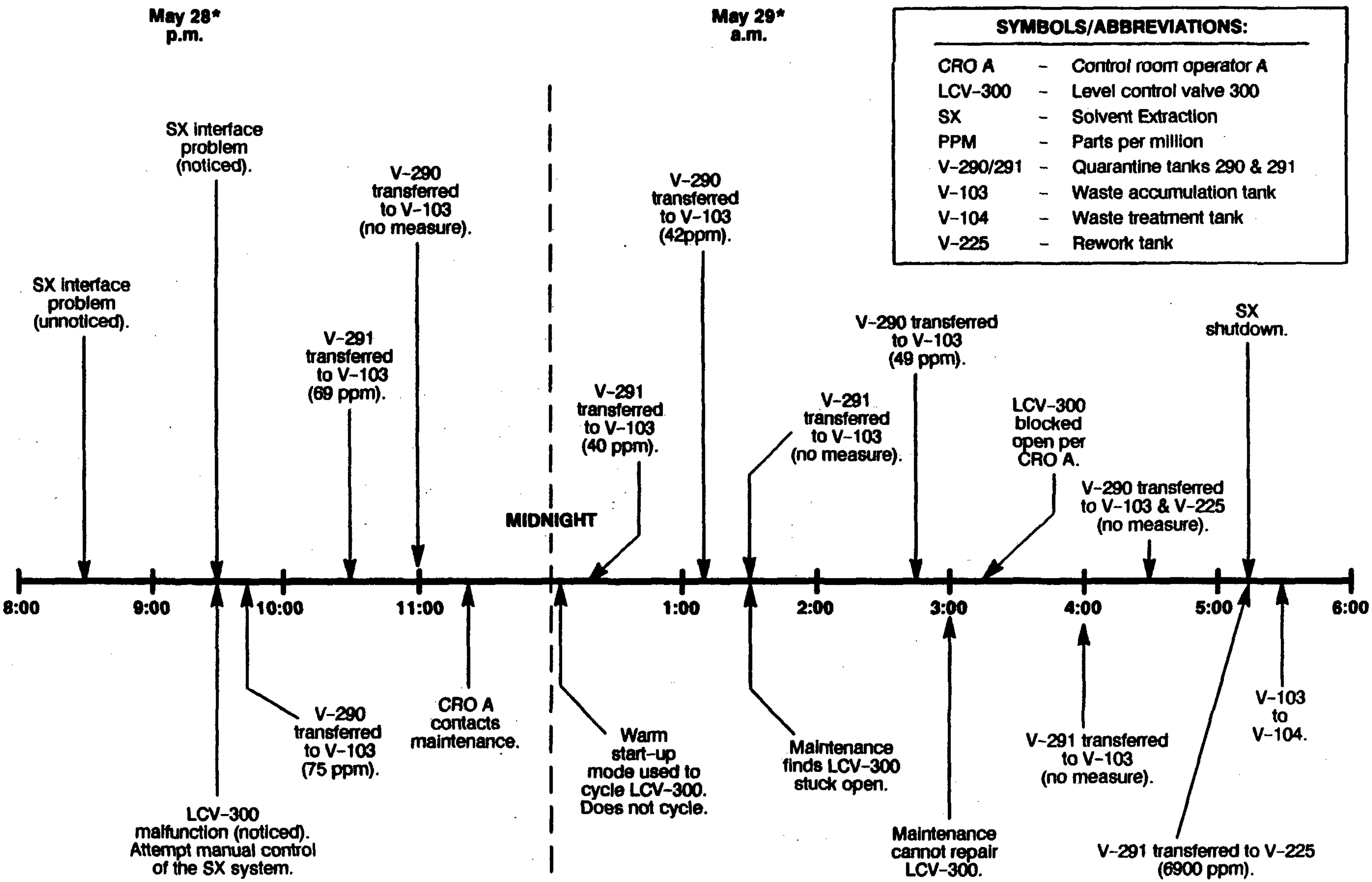


Figure 1.2 Sequence of events for the May 28–29, 1991, portion of the GE-Wilmington incident

* Sample results shown proved to be inaccurate because of sampling system deficiencies.

# 2 DESCRIPTION OF TEAM ACTIVITIES

## 2.1 Team Approach

The Incident Investigation Team (IIT) conducted its investigation in accordance with the guidance specified in the U.S. Nuclear Regulatory Commission's (NRC) Incident Investigation Manual (NUREG-1303), February 1988. The team collected and evaluated information and data from: (1) records and logs, (2) tests, (3) physical inspections and examinations, (4) written procedures, (5) system description and vendor manuals, (6) photographs, (7) system demonstrations and walkdowns, and (8) meetings and interviews. In addition to interviews of licensee personnel, the team interviewed representatives of the State of North Carolina Radiation Protection and Emergency Management Program, the New Hanover County Department of Emergency Management, as well as NRC headquarters and regional personnel.

The team quarantined the systems and equipment that failed or contributed to the incident shortly after the incident occurred to preserve information and evidence that could be used to identify root causes. After the quarantine was lifted, the licensee developed a troubleshooting plan to systematically evaluate systems and equipment. These licensee actions and plans were coordinated with the IIT. Upon completion of its field investigation at the General Electric facility in Wilmington, North Carolina, the team reassembled at NRC headquarters to analyze and evaluate the information and data it gathered.

## 2.2 Interviews and Meetings

The team arrived in Wilmington, North Carolina, on June 2, 1991, and received an incident briefing and facility overview from members of the NRC Region II Response Team who were still on site. On June 3, the IIT held an entrance meeting and conducted walkdowns of certain facility systems and areas. The IIT began conducting transcribed interviews on the afternoon of June 3 and continued through June 13. They gave priority to interviews of management, shift operations, maintenance, and laboratory personnel. Typed transcripts prepared following the interviews were made available to the team and to interviewees the following day. Personnel interviewed were allowed to review the typed transcripts and complete supplemental errata sheets for any necessary corrections and clarifications.

With the exception of the entrance and exit meetings, team meetings and status meetings between the licensee and the IIT were not transcribed. Some licensee personnel were interviewed more than once when additional information or a clarification of facts and issues was needed. Table 2.1 provides a listing of interviews and meetings that the team conducted.

## 2.3 Facility Data and Information

Operating information and data pertaining to selected process systems and components were collected for review. This information and data consisted of, but was not limited to, log entries, sample analysis result records, maintenance work histories, computer printouts, process control system computer screen displays, production reports, calibration records, quarantine tank operation records, facility change requests, radiological monitoring/survey results, incident/event reports, and aqueous waste sample data sheets. Reviews were made of NRC Headquarters Operations Center teleconference tapes and event logs and NRC regional office event logs and status summaries. In addition to this information and data, the team took approximately 200 photographs of selected process systems and components and examined approximately 400 documents during the course of the review.

Table 2.1 Interviews and meetings conducted by the GE-IIT

| Date | Time | Meeting/Interview |
|---|---|---|
| 6/3/91 | 9:00 a.m. | Entrance Meeting IIT/GE |
| 6/3/91 | 3:00 p.m. | Interview of A operator, D-Shift |
| 6/3/91 | 3:00 p.m. | Interview of laboratory technical analyst |
| 6/3/91 | 5:00 p.m. | Interview of A-shift area coordinator |
| 6/3/91 | 5:00 p.m. | Interview of A operator, A-shift |
| 6/4/91 | 9:00 a.m. | Interview of interim emergency director |
| 6/4/91 | 9:00 a.m. | Interview of A operator, B-shift |
| 6/4/91 | 10:00 a.m. | Interview of area coordinator, D-shift |
| 6/4/91 | 1:00 p.m. | Interview of waste treatment A operator |
| 6/4/91 | 2:00 p.m. | Interview of area coordinator, B-shift |
| 6/4/91 | 2:00 p.m. | Interview of principal engineer/technical leader |
| 6/4/91 | 3:00 p.m. | Interview of senior program manager |
| 6/5/91 | 7:00 a.m. | Interview of instrument technician |
| 6/5/91 | 8:00 a.m. | Interview of lab analyst |
| 6/5/91 | 8:00 a.m. | Interview of instrumentation and control technician |
| 6/5/91 | 10:00 a.m. | Interview of manager, manufacturing technology (interim emergency director) |
| 6/5/91 | 10:00 a.m. | Interview of principal engineer/technical leader |
| 6/5/91 | 11:00 a.m. | Interview of interim emergency director |

Table 2.1 Interviews and meetings conducted by the GE-IIT

| Date | Time | Meeting/Interview |
|---|---|---|
| 6/5/91 | 1:00 p.m. | Interview of lead engineer, instrumentation and control |
| 6/5/91 | 1:00 p.m. | Interview of principal engineer |
| 6/5/91 | 3:00 p.m. | Interview of FMO building manager |
| 6/5/91 | 3:00 p.m. | Interview of instrument controls technician |
| 6/5/91 | 4:00 p.m. | Interview of lead engineer, instrumentation and control |
| 6/5/91 | 4:00 p.m. | Interview of maintenance coordinator, B-shift |
| 6/5/91 | 5:00 p.m. | Interview of lab analyst, B-shift |
| 6/5/91 | 5:00 p.m. | Interview of instrumentation technician |
| 6/5/91 | 6:00 p.m. | Interview of manager, environmental protection |
| 6/5/91 | 6:30 p.m. | Interview of radiation protection shift supervisor |
| 6/6/91 | 8:00 a.m. | Interview of A operator, D-shift |
| 6/6/91 | 8:00 a.m. | Interview of manager, radiation protection |
| 6/6/91 | 9:00 a.m. | Interview of criticality safety engineer |
| 6/6/91 | 10:00 a.m. | Interview of criticality safety engineer |
| 6/6/91 | 10:00 a.m. | Interview of process engineer, fuel technology resources |
| 6/6/91 | 1:00 p.m. | Interview of manager, nuclear safety engineering |
| 6/6/91 | 1:00 p.m. | Interview of lab analyst |

Table 2.1 Interviews and meetings conducted by the GE-IIT

| Date | Time | Meeting/Interview |
|---|---|---|
| 6/6/91 | 2:00 p.m. | Interview of nuclear criticality safety engineer |
| 6/6/91 | 2:00 p.m. | Interview of FMO building manager |
| 6/6/91 | 3:00 p.m. | Interview of manager, human resources |
| 6/6/91 | 3:00 p.m. | Interview of B operator, B-shift |
| 6/6/91 | 3:30 p.m. | Interview of manager, equipment reliability |
| 6/6/91 | 4:00 p.m. | Interview of deputy team leader, GE investigation team |
| 6/6/91 | 5:00 p.m. | Interview of A operator, B-shift |
| 6/7/91 | 8:00 a.m. | Interview of program manager, fuel manufacturing operations |
| 6/7/91 | 9:00 a.m. | Interview of manager, regulatory compliance |
| 6/7/91 | 10:00 a.m. | Interview of waste treatment engineer |
| 6/7/91 | 11:00 a.m. | Interview of acting plant manager |
| 6/7/91 | 11:00 a.m. | Telephone interview of State of North Carolina radiological health officials |
| 6/7/91 | 1:00 p.m. | Interview of manufacturing engineer specialist |
| 6/7/91 | 2:00 p.m. | Interview of fuel technical resources group member |
| 6/7/91 | 3:00 p.m. | Interview of B operator, B-shift |
| 6/7/91 | 4:00 p.m. | Interview of A operator, B-shift |
| 6/8/91 | 10:00 a.m. | Meeting and status briefing on quarantine list and start up items |

Table 2.1 Interviews and meetings conducted by the GE-IIT

| Date | Time | Meeting/Interview |
|---|---|---|
| 6/9/91 | 6:00 a.m. | Interview of A operator, D-shift |
| 6/9/91 | 7:00 a.m. | Interview of B operator, D-shift |
| 6/9/91 | 8:00 a.m. | Interview of B operator, D-shift |
| 6/10/91 | 8:00 a.m. | Interview of principal engineer/technical leader |
| 6/10/91 | 8:00 a.m. | Interview of senior engineer/technical leader, instrumentation and control |
| 6/10/91 | 9:00 a.m. | Interview of URU planner |
| 6/10/91 | 9:00 a.m. | Interview of maintenance coordinator, uranium process |
| 6/10/91 | 10:00 a.m. | Interview of production maintenance manager |
| 6/10/91 | 10:30 a.m. | Interview of Director, New Hanover County Department of Emergency Management |
| 6/10/91 | 11:00 a.m. | Interview of training and development manager |
| 6/10/91 | 1:00 p.m. | Interview of manager, regulatory compliance |
| 6/10/91 | 1:00 p.m. | Interview of supervisor, HVAC maintenance |
| 6/10/91 | 3:00 p.m. | Interview of technical leader, instrumentation and control |
| 6/10/91 | 4:00 p.m. | Interview of nuclear criticality safety engineer |
| 6/10/91 | 5:00 p.m. | Interview of nuclear criticality safety engineer |
| 6/11/91 | 8:00 a.m. | Interview of A operator, A-shift |
| 6/11/91 | 8:00 a.m. | Interview of process control specialist - Met Lab |

Table 2.1 Interviews and meetings conducted by the GE-IIT

| Date | Time | Meeting/Interview |
|---|---|---|
| 6/11/91 | 9:00 a.m. | Interview of A operator, A-shift |
| 6/11/91 | 10:00 a.m. | Interview of FMO building manager |
| 6/11/91 | 10:00 a.m. | Interview of manager, nuclear safety engineering |
| 6/11/91 | 1:00 p.m. | Telephone conversation/interview of Assistant Director State of North Carolina Emergency Management Agency |
| 6/11/91 | 1:00 p.m. | Interview of A operator, C-shift |
| 6/11/91 | 2:00 p.m. | Interview of two nuclear criticality safety engineers |
| 6/11/91 | 3:00 p.m. | Interview of lab analyst, B-shift |
| 6/12/91 | 9:00 a.m. | Interview of senior engineer, plant engineering and maintenance |
| 6/12/91 | 3:00 p.m. | Interview of A operator, C-shift |
| 6/13/91 | 8:00 a.m. | Interview of manager, nuclear safety engineering |
| 6/26/91 | 8:00 a.m. | Interview of Senior Radiation Specialist, NRC Region II |
| 6/26/91 | 8:45 a.m. | Interview of Emergency Preparedness Analyst, NRC Region II |
| 6/26/91 | 10:15 a.m. | Interview of Branch Chief, Radiological Protection and Emergency Preparedness, NRC Region II |
| 6/26/91 | 11:00 a.m. | Interview of Radiation Specialist, NRC Region II |
| 6/26/19 | 12:00 p.m. | Interview of Director, Division of Industrial and Medical Nuclear Safety, NRC Headquarters |
| 6/26/91 | 1:00 p.m. | Interview of Fuel Facility Inspector, NRC Region II |

Table 2.1 Interviews and meetings conducted by the GE-IIT

| Date | Time | Meeting/Interview |
|---|---|---|
| 6/26/91 | 2:00 p.m. | Interview of Section Chief, Radiation Safety Projects Section, NRC Region II |
| 6/26/91 | 3:00 p.m. | Interview of Fuel Facility Inspector, NRC Region II |
| 6/26/91 | 4:00 p.m. | Interview of Section Chief, Emergency Preparedness Section, NRC Region II |
| 6/27/91 | 8:00 a.m. | Interview of Director, Division of Radiation Safety and Safeguards, NRC Region II |
| 6/27/91 | 9:30 a.m. | Interview of Senior Fuel Facility Inspector, NRC Region II |
| 6/27/91 | 2:30 p.m. | Interview of Section Chief, Fuel Cycle Safety Section, NRC Headquarters |
| 6/27/91 | 3:50 p.m. | Interview of Project Manager, Fuel Cycle Safety Section, NRC Headquarters |

# 3 INCIDENT NARRATIVE AND CHRONOLOGY

This section describes the sequence of events from May 28, 1991 through June 3, 1991, at the General Electric Company's Fuel and Components Manufacturing Facility that led to the inadvertent transfer of approximately 150 kilograms (320 pounds) of uranium to a waste treatment tank with an unfavorable geometry. ("Unfavorable geometry" refers to a container or vessel that can hold enough uranium to produce a criticality.) This chronology also includes the uranium recovery to a safe condition through June 3, 1991. The Incident Investigation Team created this narrative and the chronological sequence of events listed in Table 3.1 from information gathered in interviews with licensee and NRC personnel, data reviews, examinations of process control system printouts, and reviews of event and operating logs and maintenance histories.

## 3.1 Facility, Initial Conditions

On May 28, 1991, routine fuel manufacturing operations were in progress on all shifts. Waste treatment operations apparently proceeded normally through the afternoon of May 28, 1991, and consisted of routine transfers from waste accumulation tanks to waste treatment tanks located at the Waste Treatment Facility (WTF). Around 2:15 p.m. on May 28, WTF operators began pumping the contents of a waste accumulation tank (V-103) to a waste treatment tank (V-104). Both tanks, V-103 and V-104, have "unfavorable geometry" and each has a capacity of 20,000 gallons. Tank V-103 is located adjacent to the Fuel Manufacturing Operations (FMO) building and V-104 is located about one quarter mile away at the WTF. (Tanks V-103 and V-104 are shown in Figures 3.1 and 3.2 respectively.) At approximately 4:00 p.m., pumping from V-103 stopped with about 3000 gallons remaining; V-104 was filled.

## 3.2 Facility Abnormal Conditions and Incident

Around 9:30 p.m. on May 28, a Uranium Recycle Unit (URU) control room operator noted that solvent-extraction Column A had organic/aqueous phase interface and density problems. Figure 3.3 shows the solvent-extraction (SX) process. Waste process flows from the SX system are shown in Figure 3.4. The control room operator also noticed that the aqueous feed rate to Column A had increased to approximately 220 liters/hour and subsequently reset it to 180 liters/hour. The control room operator requested that the floor operator check Column A for problems. Although the operators became aware of the interface and density problems around 9:30 p.m., the problem started about an hour earlier but was apparently unnoticed by the operators. (See URU control data in Figure 3.5.) The floor operator advised the control room operator that the solvent-extraction Column A level control valve (LCV-300) was not cycling. (LCV-300 is a 1-inch globe valve having a pneumatic position actuator. Its operation is described in detail in Section 9.) The control room operator attempted to cycle the valve with commands to the process control computer. The floor operator further advised the control room operator that the valve would not respond to command signals. Although LCV-300 was malfunctioning, the solvent-extraction process continued while the floor operator attempted to control the process by throttling a manual valve located upstream of LCV-300. Figure 3.6 shows LCV-300 and the upstream manual valve, respectively.

As the solvent-extraction process continued, aqueous waste continued to be transferred to the 600-gallon favorable-geometry quarantine tanks V-290 and V-291. (These tanks are shown in Figures 3.3 and 3.4.) At about 8:50 p.m. on May 28, 1991, with tank V-290 about 62 percent full, a sample of the contents was measured and indicated 75 ppm uranium (the transfer criticality safety limit was 150 ppm uranium)[1]. Data relative to the discharges of tanks V-290 and V-291 are shown in Figure 3.5. The tank reached 90 percent full at 9:10 p.m. and the contents were transferred to tank V-103 at about 9:35 p.m.

At 10:00 p.m., a sample of the contents of tank V-291 was measured and indicated 69 ppm uranium. The tank reached 90 percent full at about 10:15 p.m. and the contents were transferred to tank V-103 at about 10:25 p.m. Both transfers to tank V-103 were done with the process control computer in the TUNE mode; that is, the automatic tank transfer function was overridden by the operator. Around 10:30 p.m., the control room operator notified the area coordinator (the on-shift supervisor) of the solvent-extraction control interface problem.

Between 10:45 p.m. and 11:00 p.m., a control room shift turnover occurred. The outgoing control room operator informed the oncoming control room operator of the problems with LCV-300 and that the upstream manual valve had been throttled. During the turnover briefing, the outgoing control room operator sampled the contents of the aqueous waste quarantine tanks and showed it to the oncoming control room operator, noting that it was "clear," an indication that excessive uranium solvent (normally a yellowish liquid) was not being transferred from the solvent-extraction columns to the aqueous waste quarantine tanks (Figure 3.4). Upon completion of shift turnover activities, the oncoming control room operator instructed the floor operator to fully reopen the throttled manual valve.

At approximately 11:00 p.m., tank V-290 reached 90 percent full and was transferred in the TUNE mode to tank V-103 (at about 11:10 p.m.) without a sample measurement. The control room operator continued to note problems with LCV-300 and solvent-extraction column interface and density. Computer data indicated that the solvent-extraction Column A interface was recovered at about 11:00 p.m. and lost again at about 11:15 p.m (Figure 3.5). At approximately 11:15 p.m., instrumentation maintenance was called and requested to look at LCV-300. Around 11:30 p.m., the control room operator called the lead process engineer at home to notify him of the solvent-extraction problem

[1] While the licensee's measurement results were recorded as a basis for decisionmaking, they proved to be inaccurate because of sampling system deficiencies.

and to obtain advice. They both agreed that putting the solvent-extraction system into the "warm startup" mode might cause LCV-300 to recycle and regain control of the solvent-extraction Column A interface. Subsequently, the control room operator attempted a "warm startup," but LCV-300 failed to respond. During this same time frame, WTF operators were decanting the contents of tank V-104 to the lagoon system. A measurement of V-104 decant samples showed the uranium content to be less than 1 ppm. Following completion of the decanting process, V-104's tank capacity was then available for subsequent discharges from waste accumulation tank V-103. At approximately 11:15 p.m., the control room operator called the floor operator to have him close the manual block valve upstream of LCV-300. Approximately 30 to 40 minutes later, the control room operator called the floor operator back to have him re-open the manual block valve.

At approximately 11:30 p.m., a sample of the contents of tank V-291 was measured and indicated 40-ppm uranium with the tank at about 50 percent full. The tank reached 90 percent full at about 12:10 a.m. on May 29, 1991, and was immediately transferred in the TUNE mode to tank V-103. Sometime around 12:15 a.m., the control room operator had the floor operator close the manual block valve again, where it remained until maintenance personnel arrived.

At approximately 1:00 a.m., a sample of the contents of tank V-290 was measured and indicated 42 ppm uranium with the tank 80 percent full. The tank reached 89.5 percent full at about 1:05 a.m. and was immediately transferred in the TUNE mode to tank V-103.

At approximately 1:30 a.m., instrumentation maintenance personnel entered the solvent-extraction area to inspect LCV-300. Subsequently, they found the valve stuck open in mid-position. When maintenance personnel removed the air supply to repair the valve positioner, LCV-300 closed as designed. During this part of the attempted repair process, verbal communications were maintained between process floor operators A and B and maintenance personnel. The control room operator, who was located in the control room remote from that part of the process floor, was unaware of the maintenance activities and valve closure. With LCV-300 closed, the uranium solvent/organic mixture interface level and density values increased and eventually flooded (i.e., backflow) the A and B solvent-extraction columns.

At about 1:35 a.m., tank V-291 reached 39 percent full and was immediately transferred in the TUNE mode to tank V-103 without a sample being measured.

Sometime between 2:30 a.m. and 3:00 a.m., instrumentation maintenance personnel completed their troubleshooting of the LCV-300 positioner. Upon testing, the valve stuck in the closed position. The positioner was disassembled again and further examination revealed that replacement parts were needed. The instrumentation maintenance personnel left the solvent-extraction area to look for replacement parts while the valve remained stuck in the closed position.

At approximately 2:55 a.m., tank V-290 reached 90 percent full and was immediately transferred in the TUNE mode to tank V-103. Although it is not clear, because of operator and chemical laboratory log inconsistencies, a sample measurement of 49 ppm of uranium is attributed to this transfer.

After further assessment, instrumentation maintenance personnel advised the control room operator that it was unlikely the valve could be repaired because of a lack of necessary repair parts. Around 3:10 a.m., instrumentation maintenance personnel, at the direction of the control room operator, blocked LCV-300 open by redirecting air pressure in the valve actuator. The flow from solvent-extraction Column A was then manually controlled by the floor operator at the valve upstream of LCV-300, as directed by the control room operator.

At approximately 4:00 a.m., tank V-291 reached 89 percent full and was immediately transferred in the TUNE mode to tank V-103 without a sample measurement. Meanwhile, instrumentation maintenance personnel rechecked the supply and parts inventory for parts necessary to repair LCV-300. Around 4:30 a.m., they reconfirmed to the control room operator that LCV-300 could not be repaired because parts were unavailable. At approximately 4:30 a.m., tank V-290 reached 64 percent full and was immediately transferred in the TUNE mode to tank V-103 and tank V-225 (a favorable geometry rework tank), without a sample measurement.

At approximately 5:10 a.m., a sample of the contents of tank V-291 was measured and indicated 6977 ppm uranium at 58 percent full. The tank reached 80.5 percent full at about 5:20 a.m., and was transferred in the TUNE mode to the V-225 rework tank. At this point, the control room operator shut the solvent-extraction process down. At 5:30 a.m., WTF operators pumped the contents of tank V-103 to tank V-104 at the WTF.

The results of sample analyses taken from tank V-104, available to WTF personnel around 7:00 a.m., showed a uranium concentration of 2333 ppm. Following notification to certain facility management personnel, the licensee decided to minimize the nuclear criticality potential by continuing air sparging (i.e., mixing with air) the tank contents to prevent the uranium precipitate from accumulating in the bottom of the tank. This action was consistent with procedural requirements which mandate sparging in tank V-104 if the uranium concentration exceeds 500 ppm. Further actions taken during this incident are detailed in the uranium recovery action section below.

## 3.3 Incident Uranium Recovery Activities

The problems associated with the solvent-extraction columns and the high uranium content in the waste treatment tank were discussed at the morning production meeting on May 29. Following the meeting, nuclear safety engineering personnel and the manager of regulatory compliance became involved in the incident. The incident was categorized as a Class II incident in conformance with the licensee's internal procedures. Based on available data, the licensee estimated that tank V-104 contained 150 kilograms of uranium. Because of the significance of the problem, the licensee assembled an evaluation/recovery team. The acting plant manager was briefed on the matter and concurred with the decision to assemble the team.

Shortly after noon, the evaluation/recovery team met to access the situation and develop remedial options. They decided that the uranium-bearing material could be removed from tank V-104 by centrifuging its contents. They also evaluated the filter system at the WTF as a means for removing the uranium. They decided not to use the system, however, because of its inefficiencies in removing suspended materials. Following this decision, the team focused its efforts on making centrifuge equipment available and making provisions for spare parts and maintenance support.

At 3:45 p.m. on May 29, the licensee advised the U.S. Nuclear Regulatory Commission (NRC) Region II office in Atlanta, Georgia, of the incident. Region II formed an initial response team and dispatched them to the site. Shortly after 5:00 p.m., licensee personnel briefed the NRC Region II and headquarters representatives concerning the incident and their ongoing corrective actions. At 5:44 p.m., NRC entered the Standby Mode because of the increased monitoring required to assess criticality safety concerns. NRC Region II then notified the State of North Carolina Radiation Protection officials at 6:30 p.m. By 7:00 p.m., the licensee was conducting centrifuge operations to recover uranium from tank V-104.

To minimize the volume of liquid nitrate waste streams to the waste process system within the URU, the licensee shut down the uranium conversion process around 7:30 p.m. Other portions of the plant, namely, powder operations, milling, pressing, and fuel rod assembly operations, were able to continue.

By 11:00 p.m., the licensee reported that five centrifuge bowls of material had been processed from tank V-104. The licensee also reported that each bowl contained about 6 kgs of total sludge material and that it took approximately 20-30 minutes to collect a bowl of centrifuge material. Collecting the material took somewhat longer than had been previously estimated because of the inherent inefficiencies associated with centrifuging material being kept in suspension (i.e., it was not allowed to settle).

At 12:55 a.m. on May 30, 1990, the NRC site team arrived at the licensee's control center and received a briefing on the situation. Following an inspection of the solvent-extraction area, tank V-103, tank V-104, and equipment at WTF, the site team briefed NRC regional and headquarters management officials. The site team then began monitoring and reviewing licensee response actions. NRC decided to provide 24-hour-a-day coverage and identified additional regional and headquarters personnel to augment the site team.

Around 6:15 a.m., the NRC provided a status update to State of North Carolina radiation protection representatives. At 6:38 a.m., the licensee reevaluated the emergency plan and procedures based on conversations with the NRC and classified the emergency as an Alert. Following the declaration, notifications were made to State and local governmental authorities.

The Alert declaration did not have an impact on the licensee's staffing levels since the licensee already had provided technical, operations, and management coverage for the incident equivalent to that called for under a declared emergency condition.

For the remainder of May 30, removal of uranium sludge by centrifuging continued. During the late evening hours, the centrifuge experienced clutch and bearing problems and was taken out of service. By 12:10 a.m. on May 31, the licensee estimated that they had removed about 285 kgs of sludge and that this quantity contained about 15 percent uranium by volume. Through the remainder of the morning hours, the centrifuge continued to experience mechanical problems. The licensee then decided to make a second centrifuge available for service.

In addition to the centrifuge problems, the licensee and NRC continued to address problems associated with inconsistent sample measurements for tank V-104. By 11:00 a.m., the second centrifuge was in service and both centrifuges were operating. By 2:00 p.m., approximately 330 kgs of sludge, containing about 56 kgs of uranium, had been removed. Sample measurement results available in this time frame showed better consistency and more reliable results because of revised sampling procedures instituted earlier in the day.

The licensee and the NRC continued to discuss options relative to prompt remediation of the nuclear criticality safety problem. The primary issues discussed included continuing to centrifuge the sludge, adding soluble poisons to tank V-104, and dividing the tank contents so that they were below the critical mass in any one tank. Following these discussions, the licensee continued centrifuge operations but made arrangements for procuring soluble poisons and made preparations for dividing the contents of tank V-104 in the WTF tankage.

Centrifuging operations continued to reduce the concentration of uranium in tank V-104 through the evening of May 31. Late in the evening, the licensee decided to transfer a portion of the contents of tank V-104 to adjacent storage tanks V-109A and V-109B (Figure 3.7). The tanks are also shown schematically on Figure 3.4. The NRC concurred with the transfer process, which commenced around 10:35 p.m. with operators using portable air-operated diaphragm pumps to transfer the contents of tank V-104 to tank V-109B. Figure 3.8 shows the pumping arrangement and the top portion of tank V-104. By 2:15 a.m. on June 1, 1991, the pumping operation was complete. However, at 2:50 a.m., it was noted that the contents of tank V-109B were being transferred to tank V-109A because of a leaking valve. By 3:15 a.m., tanks V-109A and V-109B held equivalent volumes because of the leaking valve. Although there was valve leakage between the tanks, there was no leakage out of the tanks. The licensee estimated the total amount transferred to be approximately 7400 gallons.

At 3:20 a.m. on June 1, 1991, the Alert condition was terminated. The NRC returned to the normal mode and communications through the Emergency Notification System (ENS) line were terminated. The NRC team remained on site to monitor further uranium recovery actions. The licensee continued to remove uranium using the centrifuges through the morning of June 3. When the last batch of sludge was processed, the licensee collected approximately 136 kgs of the estimated 150 kgs of uranium originally transferred.

Table 3.1 Chronological Sequence of Events

| Date | Time (EDT) | Description of Event |
|---|---|---|
| Initial Conditions | | |
| 5/28/91 | 2:15 p.m. | The pumping of waste tank V-103 to tank V-104 was initiated. Tank V-103 is located adjacent to the Uranium Recycle Unit (URU) and tank V-104 is located at the Waste Treatment Facility (WTF). V-103 and V-104 are 20,000-gallon unfavorable geometry tanks. |
| | 4:00 p.m. | Pumping from V-103 ceased with only approximately 3,000 gallons remaining and with tank V-104 filled. Records indicate a transfer of 15,600 gallons. A measured sample of tank V-104 contents indicated 99.2 ppm uranium (U).<br><br>NOTE: The waste streams subsequently transferred into tank V-103 contained the material transferred to tank V-104 that was found to contain an unauthorized amount of uranium. |
| First Indication of Abnormalities | | |
| | 8:30 p.m. | Interface problems begin to occur with the solvent-extraction (SX) column (as evidenced by historical data). Operators are apparently unaware at this point. |
| | 9:30 p.m. | Control Room Operator A (CRO A) noted that SX Column A had interface and density problems. The process floor operator indicated that the SX Column A level control valve (LCV-300) was not actuating and did not respond to demand signals. The LCV-300 is a 1" globe valve with a pneumatic positioner actuator. SX operations continued by throttling a manual valve adjacent to LCV-300. |

Table 3.1 Chronological Sequence of Events

| Date | Time (EDT) | Description of Event |
|---|---|---|
| | 9:35 p.m. | Aqueous Waste (AW) Quarantine (Q) tank V-290 at the 90-percent level was transferred to V-103. A measured sample of V-290 contained 75 ppm U[2]. Release limit from V-290 to V-103 is 150 ppm U. V-290 is a 600-gallon criticality safe geometry tank. |
| | 10:25 p.m. | Q tank V-291 at the 90-percent level was transferred to V-103. A measured sample of V-291 indicated 69 ppm U. The release limit from V-291 to V-103 is 150 ppm U. V-291 is a 600-gallon criticality safe geometry tank. |
| | 10:45 p.m. - 11:00 p.m. | Control room shift turnover. |
| | 11:00 p.m. | Q tank V-290 at the 90-percent level was transferred to V-103. The contents were transferred without a sample measurement.<br><br>The relief CRO noted SX Column A interface problems.<br><br>NOTE: Records show that SX column interface was recovered for a short period in this time frame. |
| | 11:15 p.m. | CRO A calls instrumentation maintenance to troubleshoot and repair LCV-300. Records show that SX interface is lost in this time frame.<br><br>CRO A instructs a floor operator to close the manual block valve upstream of LCV-300. |
| | 11:30 p.m. | CRO A calls the process engineer to notify him of SX control problems. The process engineer concurs with CRO A to place the SX system in "warm startup" mode to attempt to get LCV-300 to cycle and position correctly. |

[2] While the licensee's measurement results were recorded and used as a basis for decisionmaking, they proved to be inaccurate because of sampling system deficiencies.

Table 3.1 Chronological Sequence of Events

| Date | Time (EDT) | Description of Event |
|---|---|---|
| | ~11:45 p.m. | CRO A instructs a floor operator to reopen the manual block valve upstream of LCV-300. |
| | 11:55 p.m. | Decant from waste neutralization tank V-104 is transferred to a lagoon. Analysis of the decant indicated < 1 ppm U. |
| 5/29/91 | 12:00 a.m. | The SX "warm-up" mode attempt does not affect LCV-300 valve position. |
| | 12:10 a.m. | Q tank V-291 was transferred at the 90-percent level to V-103. A measured sample of V-291 indicated 40 ppm U. The release limit is 150 ppm U. |
| | 12:15 a.m. | SX column interface is regained. CRO A instructs a floor operator to close the manual block valve upstream of LCV-300. |
| | 1:05 a.m. | Q tank V-290 at 89.5 percent full was transferred to V-103. A measured sample of 42 ppm U is attributed to tank V-290. The release limit is 150 ppm U. |
| | 1:35 a.m. | Instrumentation technicians found LCV-300 stuck in the open position. Troubleshooting/repair activities begin. When the air supply to the valve was removed, the valve closed, as designed. |
| | | Q tank V-291 is at the 39-percent level and is discharged to V-103. Contents were not sampled and measured for uranium concentration. |
| | 2:30 a.m. | Instrumentation personnel continue to troubleshoot the valve positioner on LCV-300 and search for repair parts. |
| | 2:55 a.m. | Q tank V-290 at 90 percent full was transferred to V-103. A measured sample of 49 ppm U is attributed to tank V-290. The release limit is 150 ppm U. |

Table 3.1 Chronological Sequence of Events

| Date | Time (EDT) | Description of Event |
|---|---|---|
| | 3:00 a.m. | Instrumentation informs CRO A that the valve cannot be repaired due to unavailability of parts. CRO A believes that SX Column A is overflowing into Column B. |
| | 3:10 a.m. | At the request of CRO A, instrumentation technicians blocks valve LCV-300 open by redirecting air pressure to the valve actuator. The AW flow from SX Column A is operated manually by the process floor operator. |
| | 3:15 a.m. | SX interface lost. |
| | 4:00 a.m. | Q tank V-291 at 81 percent full was transferred to V-103 without a sample measurement. |
| | 4:30 a.m. | Instrumentation personnel re-confirm to CRO A that LCV-300 cannot be repaired because of parts unavailability. |
| | | Q tank V-290 at 64 percent full is transferred (a split between to V-103 and V-225 [a favorable geometry rework tank]). The contents were not measured for uranium concentration. |
| | 5:20 a.m. | Q tank V-291 at 80.5 percent full was transferred to the safe geometry rework tank V-225. A measured sample of V-291 indicated 6977 ppm U. |
| | 5:20 a.m. | CRO A shuts down the SX process. |
| | 5:30 a.m. | The contents of tank V-103 are pumped to tank V-104 at the Waste Treatment Facility. |

Table 3.1 Chronological Sequence of Events

| Date | Time (EDT) | Description of Event |
|---|---|---|
| | 7:00 a.m. | A sample taken from tank V-104 indicated 2333 ppm U. |
| | | NOTE: As a result of the high V-104 sample, GE management notifications were made and a technical evaluation team was convened. In addition, tank V-104 sparging was maintained in an effort to control density by keeping the uranium precipitate suspended. |
| Uranium Recovery Activities | | |
| | 3:45 p.m. | The licensee advised Region II of the event. |
| | 5:15 p.m. - 5:30 p.m. | An NRC initial site team is assembled and dispatched. |
| | 5:44 p.m. | NRC enters Standby Mode; Region II and Headquarters Incident Response Center are activated to monitor the situation. |
| | 6:30 p.m. | The licensee provides a status briefing and action plan to NRC Headquarters and Region II. |
| | | Region II notifies the State of North Carolina Radiological Protection authorities. |
| | 7:00 p.m. | The licensee initiates uranium recovery operations from tank V-104. |
| | 8:30 p.m. | NRC Commissioner Assistants are briefed on the situation. |
| | 9:15 p.m. | NRC completes notification of appropriate Federal agencies and other regions. |
| 5/30/91 | 12:55 a.m. | A Region II site team arrived at GE Wilmington, North Carolina. The licensee provided a briefing on the situation and actions being taken. |

Table 3.1 Chronological Sequence of Events

| Date | Time (EDT) | Description of Event |
|---|---|---|
| | 2:00 a.m. | The NRC site team conducts an inspection of SX activities and the WTF. Verification is made concerning sparging of tank V-104 and backup sparging provisions. |
| | 4:00 a.m. | The NRC Site Team briefs the Region and Headquarters on facility status. |
| | 6:38 a.m. | The licensee declares an Alert. |
| | 8:30 a.m. | U-235 enrichment in V-104 is determined to be 3.125 percent (3.2% considering measurement uncertainty). Sludge removal continues. |
| | 6:30 p.m. | Region II briefs a radiological protection representative from the State of North Carolina. |
| 5/31/91 | | Centrifuge operations continued throughout the day to reduce the uranium mass in V-104 below the critical limit. Centrifuge equipment problems are experienced. |
| 6/1/91 | 3:15 a.m. | The contents of V-104 continue to be reduced by use of a centrifuge. The transfer of a portion of the contents of tank V-104 to tanks 109A and 109B is completed. A less-than-critical mass in each of the three tanks is achieved. |
| | 3:20 a.m. | The Alert is terminated by the licensee. |
| | | NRC went out of standby mode. |
| 6/2/91 | | Centrifuge operations continue. |
| 6/3/91 | ~10:30 a.m. | Centrifuge operations are complete. The licensee collected approximately 136 kgs of the estimated 150 kgs of uranium originally transferred. |

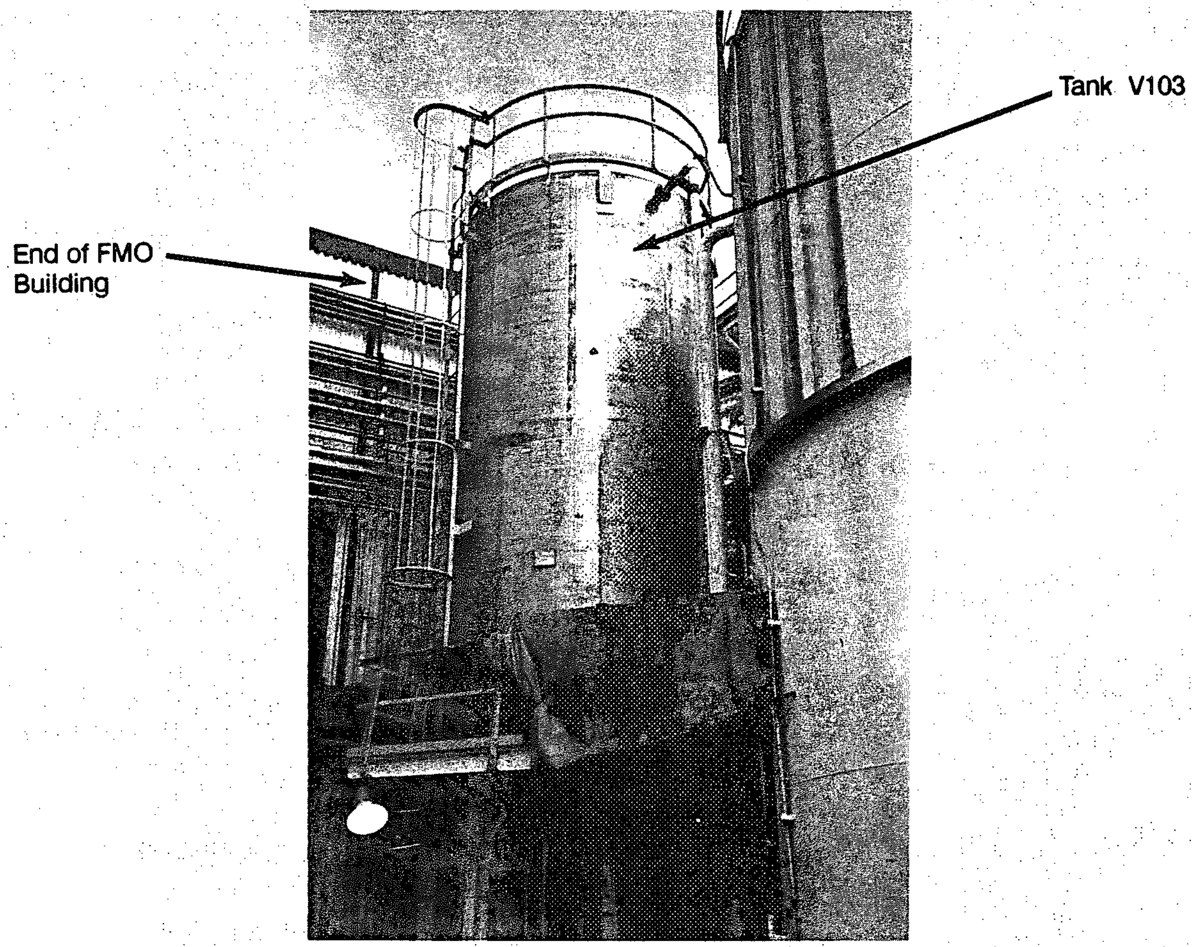


Figure 3.1 Tank V–103

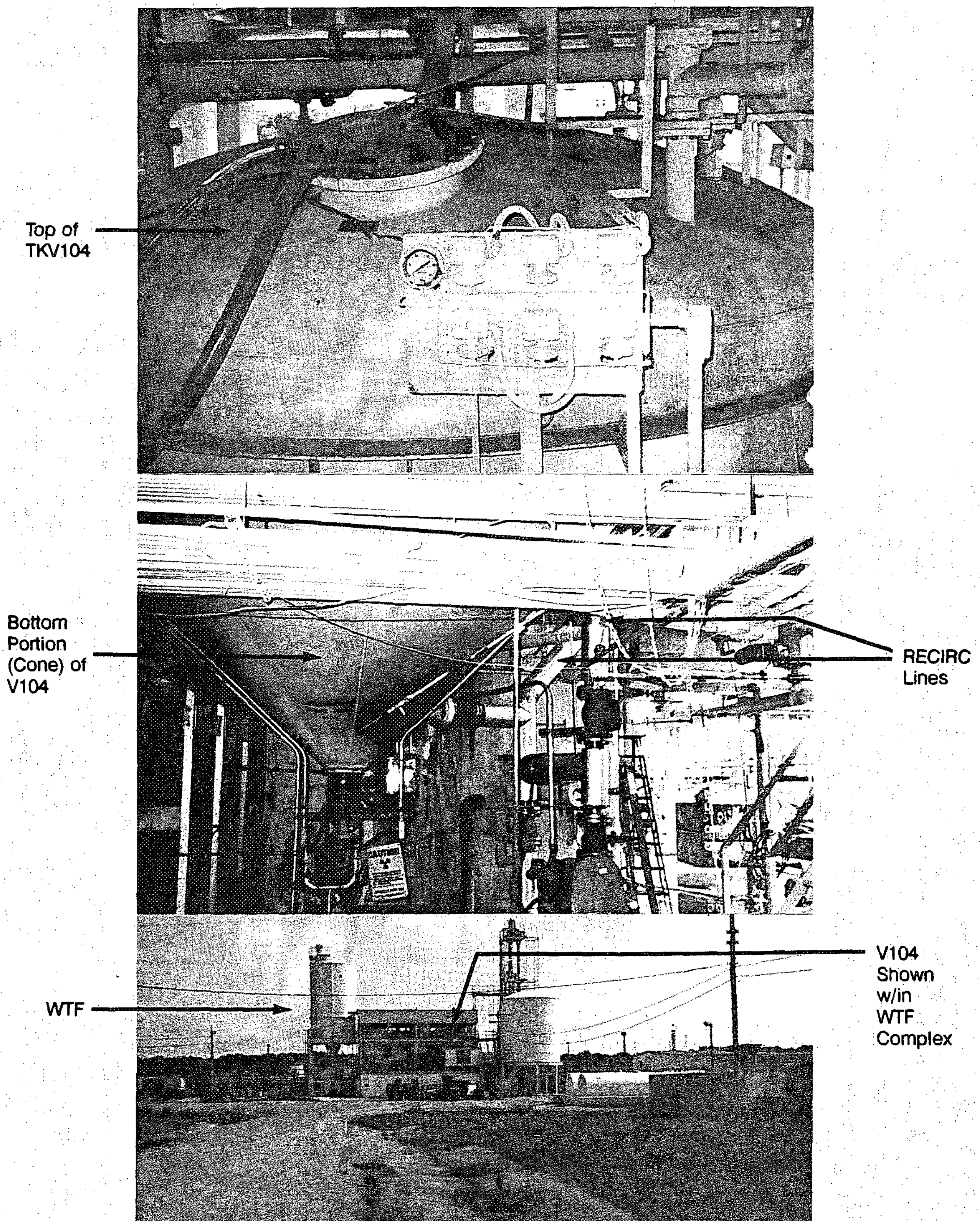


Figure 3.2 WTF and Tank V–104

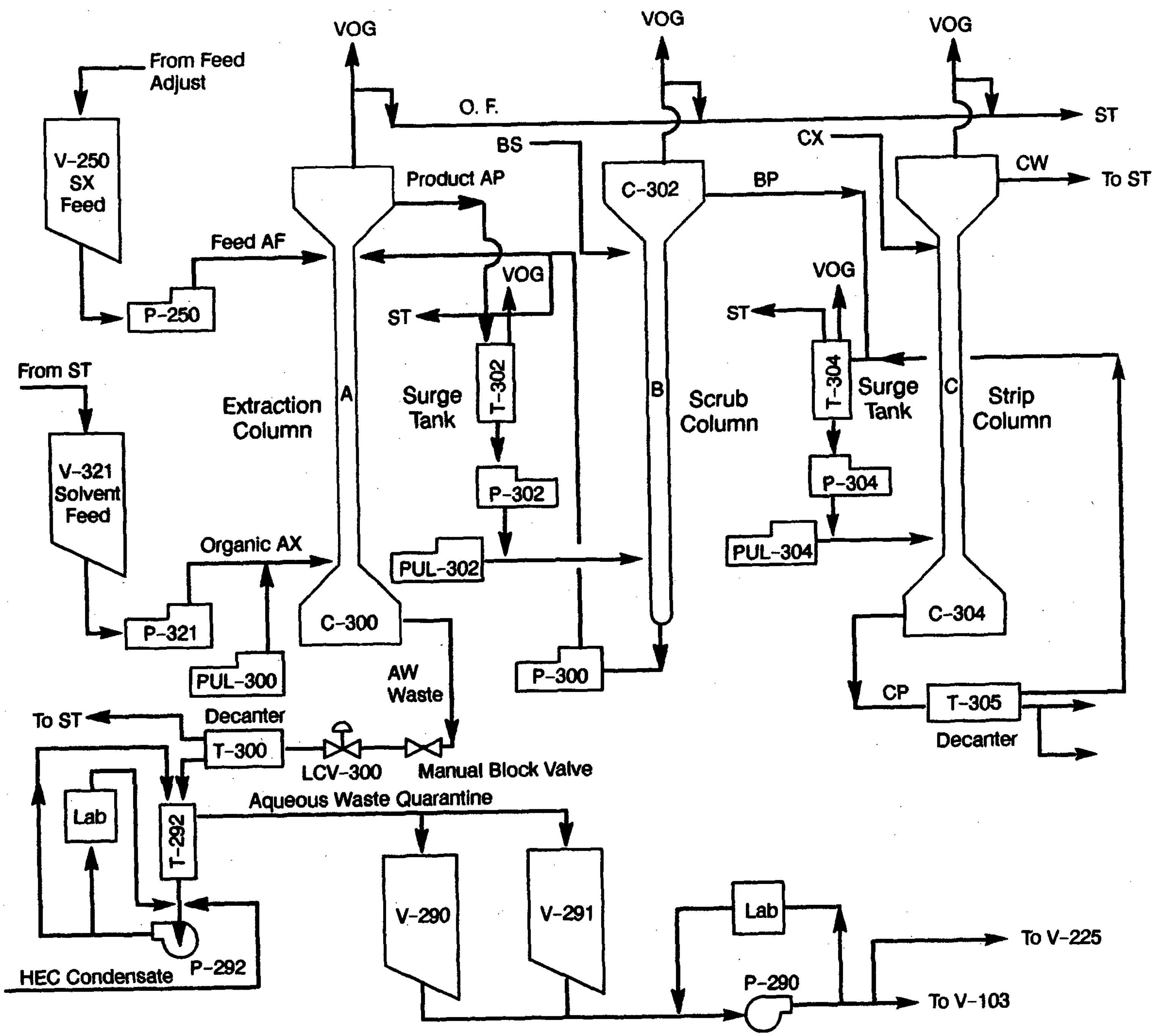


Figure 3.3 Solvent-extraction process

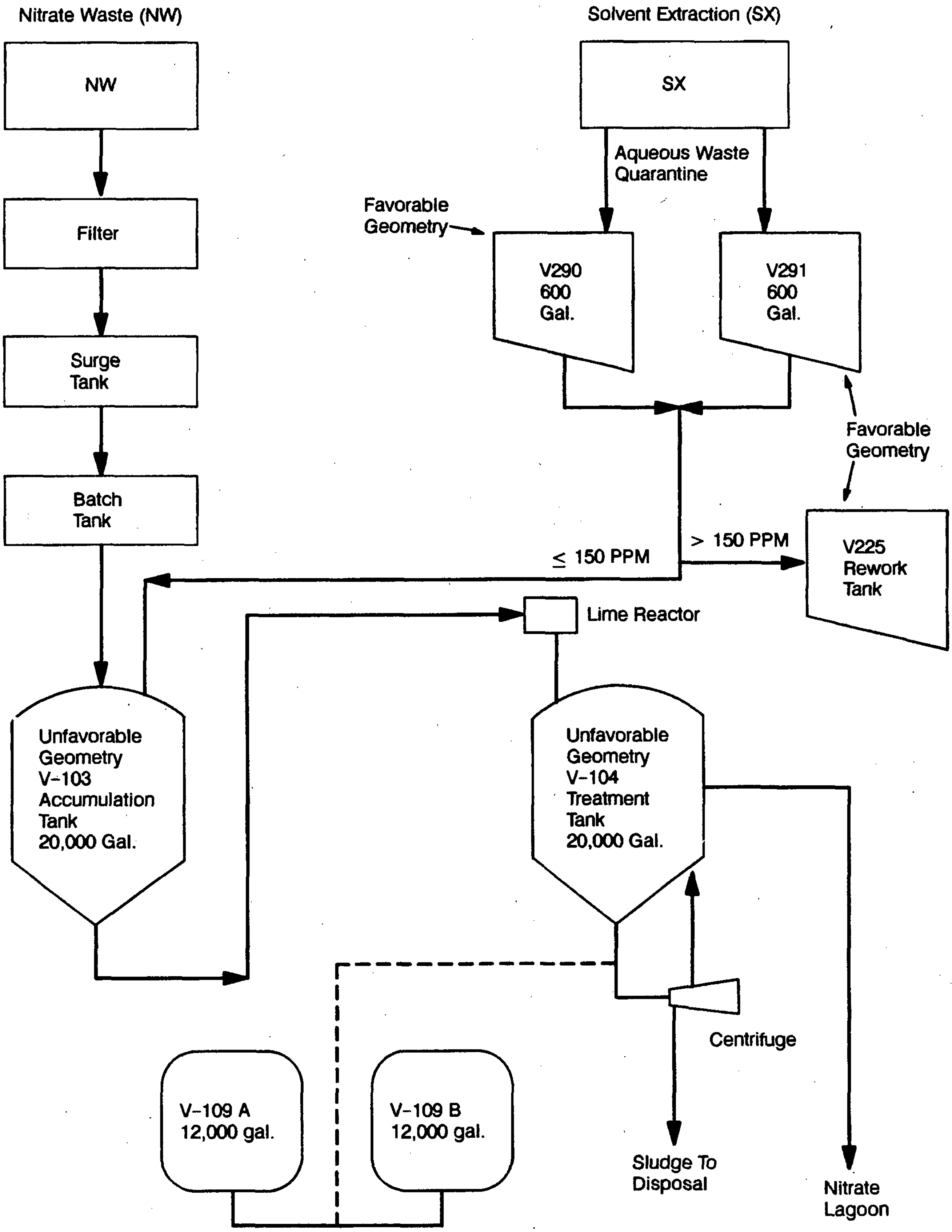


Figure 3.4 Nitrate waste and solvent-extraction process waste flow

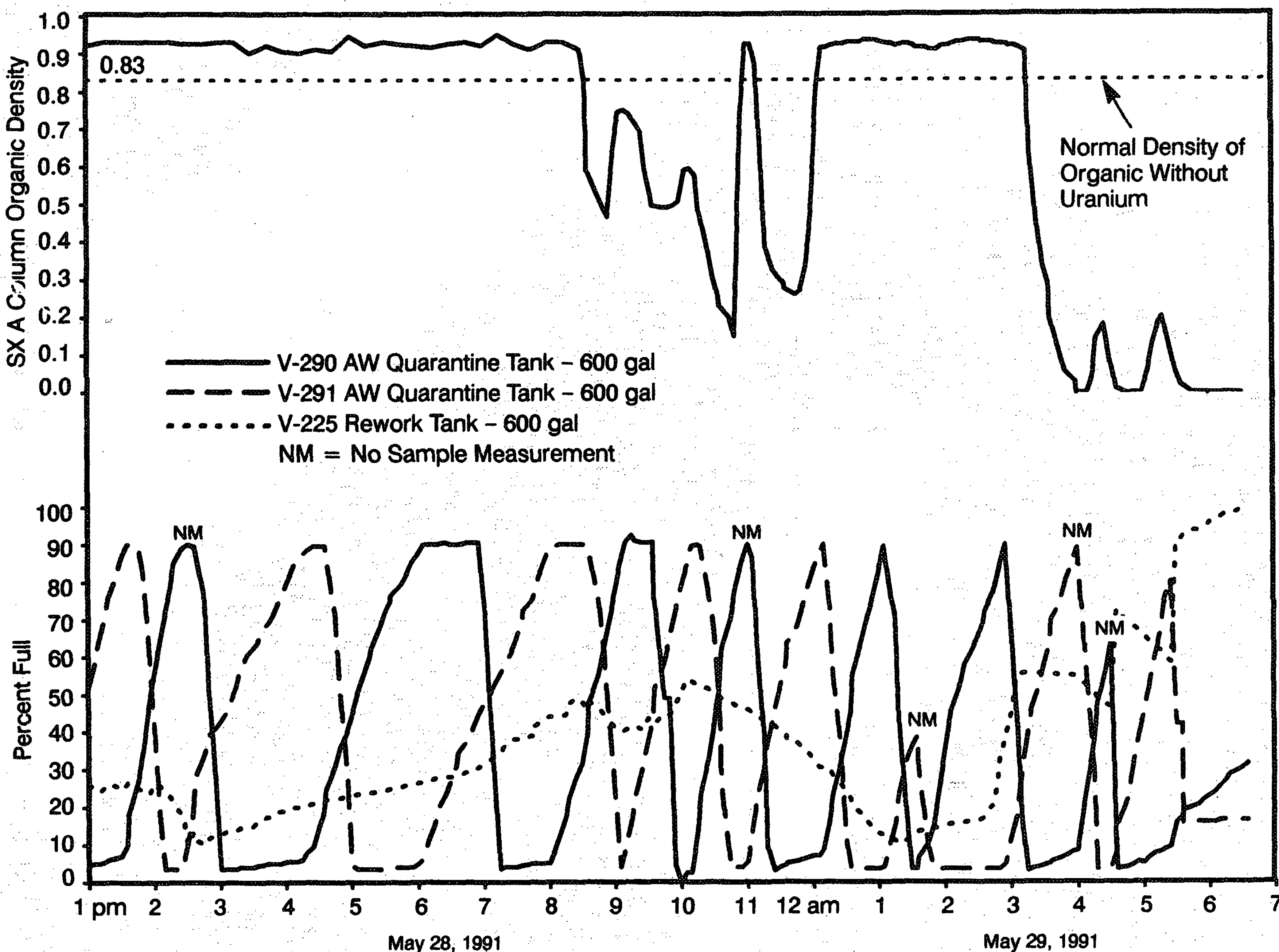


Figure 3.5 Uranium Recycle Unit process control data

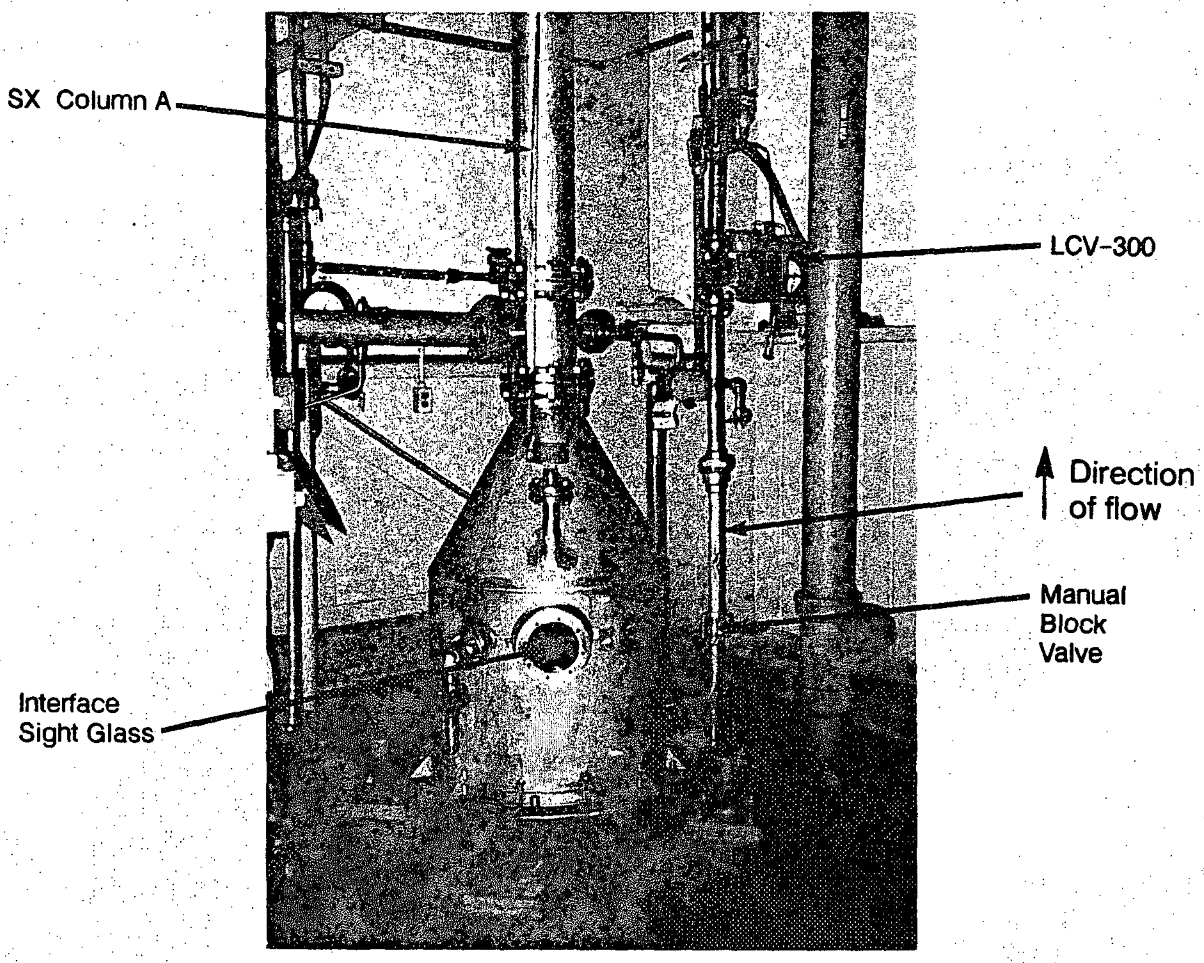


Figure 3.6 LCV-300 and Upstream Manual Valve

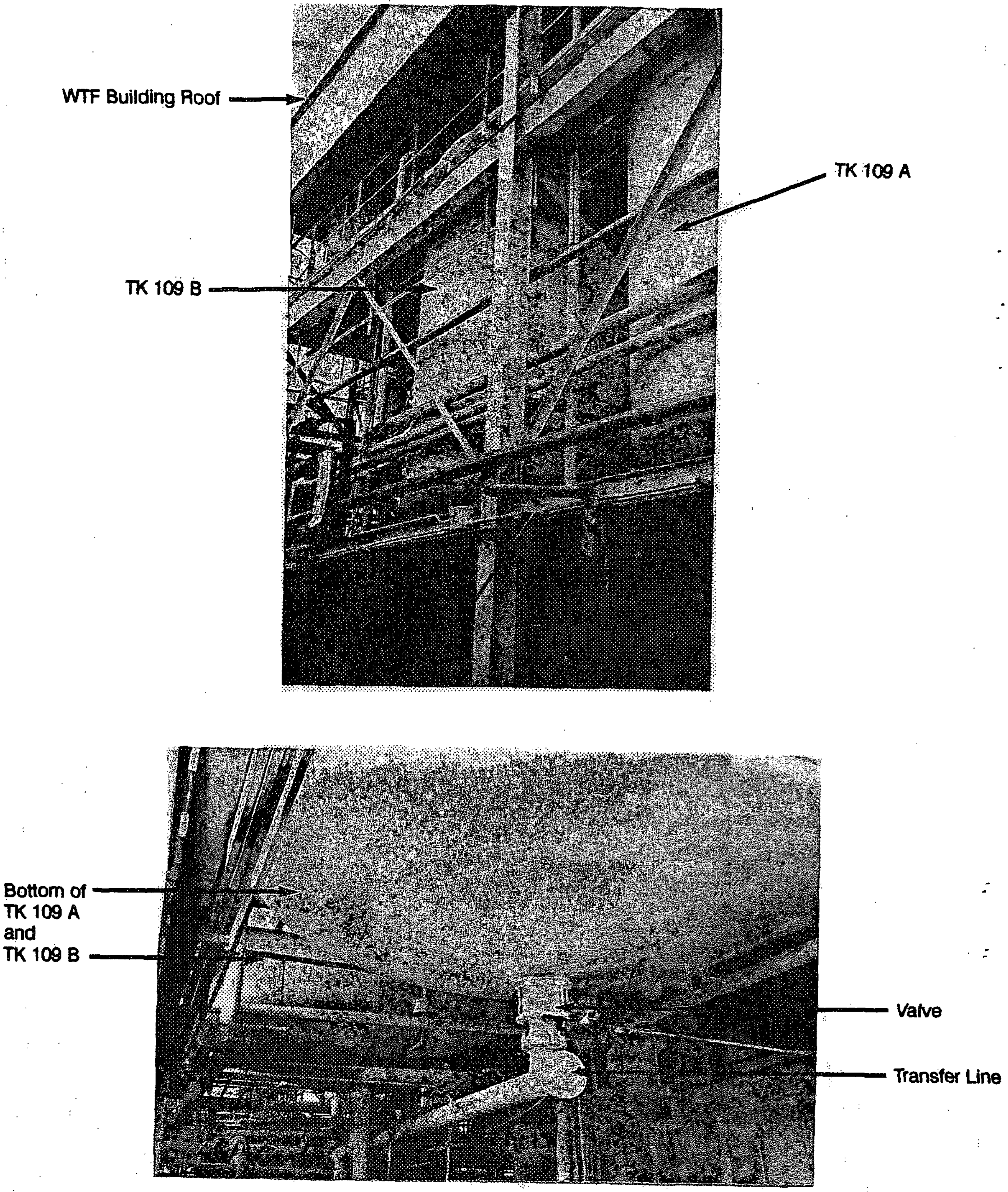


Figure 3.7 Tanks 109A and 109B

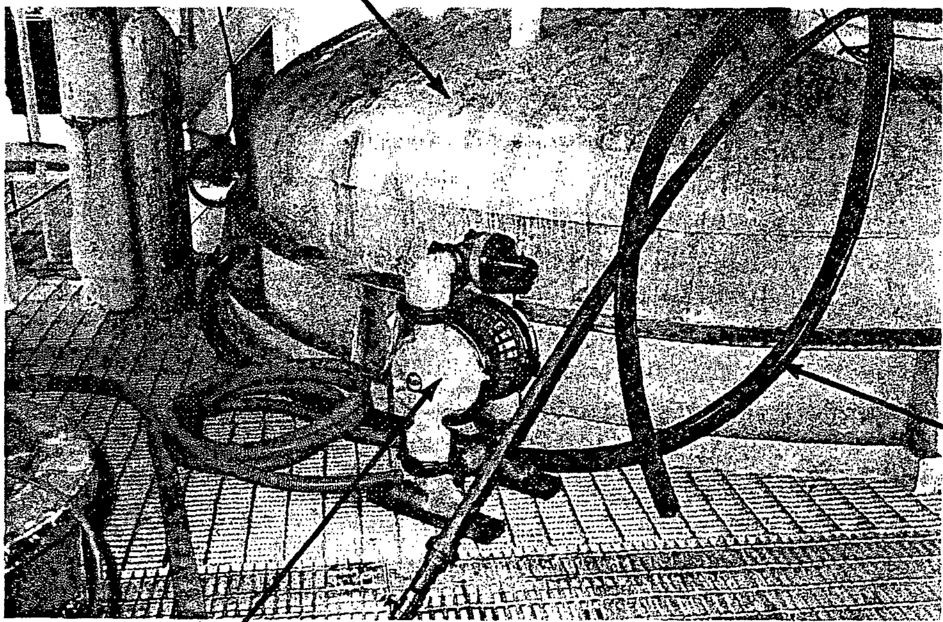


Figure 3.8 Transfer pump and top of tank V–104 (shown disconnected)

# 4 FACILITY DESCRIPTION AND PROCESS OPERATIONS

## 4.1 Facility Description

The General Electric Nuclear Fuel and Component Manufacturing Operation facility is located about six miles north of Wilmington, North Carolina (Figure 4.1). The 1600-acre site includes several separate manufacturing operations. Figure 4.2 shows the layout of site facilities. The principal operations on the site include fuel manufacturing operations, fuel component operations, service component operations, aircraft engine operations, and waste treatment facilities. Fuel manufacturing operations, the only activity licensed by the U.S. Nuclear Regulatory Commission (NRC) on site, involve fuel manufacturing and include uranium hexafluoride ($UF_6$) conversion, fuel fabrication, scrap recovery, certain process technology operations, and laboratory operations. Fuel component operations involve the manufacture of non-nuclear components for reactor fuel assemblies. Service component operations involve the manufacture of non-radioactive nuclear reactor components. Aircraft engine operations involve the manufacture of commercial aircraft engines. Finally, waste treatment facilities process certain liquid waste streams generated elsewhere on the site as a result of processing operations. Depending on the nature of the wastes (fluoride or nitrates), they are treated and/or processed either for discharge to the lagoon system, for disposal as solid material, or for recycling. A flowchart depicting the entire fuel manufacturing process is shown in Figure 4.3. Uranium recovery operations are shown in the shaded area of Figure 4.3.

This section of the Incident Investigation Team report describes uranium recovery operations, the process involved in the incident. The manufacturing processes that generate the waste streams and internally generated uranium scrap that comprise the materials fed to uranium recovery operations are described in detail in Appendix C.

The upset condition that initiated this incident occurred in the solvent-extraction process, which is described in Section 4.2.7.3 below.

## 4.2 Uranium Recovery

The uranium recovery process recovers uranium in the form of uranyl nitrate from waste streams and internally generated uranium scrap. (The uranium recovery process is shown in Figure 4.4.) Each major step in the uranium recovery and waste treatment process is described below.

### 4.2.1 Fluoride Waste Treatment - Uranium Recovery

The chemical conversion of uranium hexafluoride to uranium dioxide produces an ammonium fluoride waste liquid. This fluoride waste liquid, normally containing low concentrations of uranium, is treated by an ion-exchange process to recover the uranium. (A flowchart of the fluoride waste treatment process is presented in Figure 4.5.)

In the fuel manufacturing building, the waste liquid is released from quarantine tanks to a 65,000-gallon surge tank which feeds the ion-exchange process. The ion-exchange process consists of three columns: two operating in series and the third, a regenerated column, in standby. Fluoride liquid flows through the two columns with the uranium being removed by the ion-exchange resin. The liquid, which normally is nearly uranium-free, flows to a 100,000-gallon surge tank at the waste treatment plant for additional processing and ammonia recovery before final discharge to the final process lagoons. An on-line sampling and measurement system monitors the uranium in the liquid being discharged to the waste treatment plant.

### 4.2.2 Fluoride Waste Treatment - Ammonia Recovery/Waste Discharge

Fluoride waste discharged from the uranium recovery ion-exchange process is treated at the fluoride waste treatment plant to recover ammonia for return to the fuel manufacturing operation and to precipitate calcium fluoride for off-site disposal. This stage of the fluoride treatment system is shown in Figure 4.5.

Fluoride waste is accumulated in a 100,000-gallon surge tank and preheated in the shell side of a vertical condenser. It is then pumped to a reactor tank where it reacts with lime in an exchange reaction to precipitate calcium fluoride ($CaF_2$) and ammonium hydroxide ($NH_4OH$). The reaction products are pumped to a steam stripping column where ammonium hydroxide is removed as an overhead product and recovered by condensing in a vertical condenser.

The aqueous calcium fluoride solution, the stripper bottom product, is mixed with a flocculant and settled in an overflow clarifier. Clear effluent from the clarifier is gravity fed to a fluoride lagoon system for additional settling before it is pumped to the final process lagoon system for discharge. The solids from the clarifier are separated by filtration and prepared for off-site disposal. The filtrate is returned to the clarifier.

### 4.2.3 Radwaste Treatment - Uranium Recovery

The radwaste treatment system is an accumulation of various low-level radioactive waste streams which are generated in the fuel manufacturing operation. The sources of these waste streams include liquids from the Chemet laboratory and the decon and waste oxidation-reduction facilities, as well as from equipment cleaning for maintenance and general housecleaning. (A flowchart of the radwaste treatment system is shown in Figure 4.6.)

The radwaste liquids are collected in two accumulation vessels within the uranium recovery operation. Other radwaste liquids generated within the uranium recovery operation are also collected in these vessels. Water from the laundry operation, where controlled area protective clothing is washed, normally bypasses uranium recovery because of its very low uranium content.

The accumulated liquids are treated with lime to precipitate the uranium, then held in aging tanks. The uranium is concentrated into a sludge by recirculating the treated liquid through a filter system. A controlled volume of slurry from the filter system is diverted to a centrifuge to remove solids. The uranium-bearing solids from the centrifuge are sent to the scrap recovery operation to recover the uranium. The filtrate, which is nearly uranium-free, is discharged through a filtrate receiver tank and a filtrate hold-up tank to the final process lagoon system.

### 4.2.4 Radwaste Treatment - Final Process Lagoon System

Radwaste is combined with other liquid effluents in the final process lagoon system. The pH of the liquid effluents is monitored and adjusted, if necessary, before final discharge from the site to the Northeast Cape Fear River. (A flowchart of the radwaste treatment-final process lagoon system is shown in Figure 4.7.)

### 4.2.5 Nitrate Waste Treatment - Uranium Recovery

The nitrate waste treatment system is an accumulation of radioactive waste streams that are generated primarily from uranyl nitrate conversion and acid flushing of process equipment. (A flowchart of the nitrate waste treatment system is shown in Figure 4.8.)

The nitrate waste liquids from uranyl nitrate conversion and acid flushing are released from quarantine tanks in the fuel manufacturing operation to an accumulation tank in the uranium recovery operation. Other nitrate waste liquids, which are generated in the uranium recovery operation, are also collected in the accumulation tank.

The accumulated liquids are treated with lime in a reactor to precipitate the uranium, then retained in an aging tank. The uranium is concentrated into a sludge by recirculating the treated liquid through a filter system. A controlled volume of slurry from the filter system is diverted to a centrifuge to remove solids. The uranium-bearing solids from the centrifuge are sent to the scrap recovery operation to recover the uranium. The filtrate, which is nearly uranium-free, is discharged to a 20,000-gallon surge tank.

The solvent-extraction process of uranium recovery generates a nitrate aqueous waste. This waste liquid, intended to be low in uranium, contains the various impurities (primarily metal impurities) rejected by the solvent-extraction process. The solvent-extraction aqueous waste is accumulated in either of two quarantine tanks in the uranium recovery area. After analytical confirmation of low uranium content, the aqueous waste is discharged to the same 20,000-gallon surge tank. The contents of the 20,000-gallon surge tank are pumped to the secondary nitrate waste treatment operation at the waste treatment plant in a batch transfer mode.

### 4.2.6 Secondary Nitrate Waste Treatment

The liquid effluents from the primary nitrate waste treatment and aqueous wastes from the solvent-extraction process are accumulated in a 20,000-gallon surge tank in the uranium recovery area, and are transferred to a settling tank at the secondary nitrate waste treatment facility in a batch mode. The waste stream to the settling tank is mixed with lime, as required, in an agitated reactor to adjust pH. The liquid is gravity fed to a draft tube in the settling tank to direct solids to the bottom of the cone. The concentrated solids, mostly metal hydroxides and residual uranium, are removed from the bottom of the tank and filtered. The filtered solids are collected and dried in the uranium recovery oxidation furnace before they are shipped for off-site disposal.

The filtrate is returned to the settling tank. The clear effluent from the settling tank is decanted through a stand pipe and gravity fed to a dedicated lagoon system for further settling. After analytical confirmation of low uranium content, the clear nitrate effluent is shipped for off-site disposal.

### 4.2.7 Scrap Recovery

Internally generated uranium-bearing scrap that does not meet quality standards or has been contaminated with foreign material is recycled through the scrap recovery operation for uranium recovery and purification. (Figure 4.9 shows the typical steps for the scrap recovery process.) The scrap recovery operation is located within the URU of the fuel manufacturing operation. The output product from the scrap recovery operation is uranyl nitrate. The processing of uranyl nitrate is discussed in Section 4.3 below.

#### 4.2.7.1 Oxidation, Dissolution, and Leaching

All uranium-bearing scrap materials, with the exception of ash from the waste oxidation-reduction facility, are oxidized in a muffle furnace. Materials leaving the furnace are processed through a "delumper" and roll crusher and then to a blending step. After blending, each container is weighed and scanned for isotopic content. The material is then normally placed in storage prior to further processing.

Oxidized scrap materials, along with ash from the waste oxidation-reduction facility, are dissolved in nitric acid to produce a crude uranyl nitrate solution. The solution is then filtered and routed to the solvent-extraction system.

Uranium scrap materials from the fuel manufacturing and uranium recovery operations that contain high amounts of insoluble solids are leached, rather than dissolved, in nitric acid solution to produce a crude uranyl nitrate solution. This solution is filtered and routed to the solvent-extraction system.

#### 4.2.7.2 Head-End Concentrator

Liquid waste streams containing low-level concentrations of uranium produced in the scrap recovery process are accumulated in a leach water surge tank. After thorough recirculation to achieve good mixing, the feed stream is introduced into a head-end concentrator composed of a reboiler unit, a slab vapor/liquid disengaging chamber, overhead condenser, and bottoms cooler.

Upon start-up of the concentrator, the concentrated solution is recycled back to the feed tank until the density of the solution meets a prescribed target density and then is diverted to the dissolver/leacher product tanks. When the concentrator reaches steady-state conditions (bottoms output density meets target density), the recycle valve to the feed tank closes and the concentrated material in the bottom is diverted to the dissolver/leacher product tanks.

Overhead steam condensate from the condenser drains by gravity to aqueous waste recirculation tank V-292. The reboiler steam condensate is also discharged to tank V-292.

#### 4.2.7.3 Scrap Recovery - Solvent-Extraction Process

The solvent-extraction process is used to remove dissolved impurities from the uranyl nitrate (UNH) solutions from the scrap recovery-dissolution, leaching, and head-end concentrator processes. (A flowchart of the solvent-extraction process is shown in Figure 3.3.) The resulting stream of dilute uranyl nitrate solution is processed in the product concentrator prior to conversion to uranium dioxide ($UO_2$). The solvent-extraction aqueous waste stream, which normally contains metal impurities, nitric acid, and low levels of uranium, is processed at secondary nitrate waste treatment. The solvent used in the solvent-extraction process is treated periodically to regenerate its properties.

Uranyl nitrate solutions for the scrap recovery-solvent-extraction process are collected in two surge tanks and transferred, as needed, to two feed adjustment tanks that operate in parallel. Feed is adjusted in batches where nitric acid and deionized water are added as needed.

The solvent-extraction process equipment incudes an extraction column, scrub column, and stripping column. Level-controlled intermediate surge tanks are also used for solvent flow control. The functions of the three columns are to extract the bulk of the uranyl nitrate from the feed stream, scrub the uranium-bearing solvent with water, and then strip the uranium back into the aqueous phase. The solvent-extraction process is controlled automatically by the process computer control system.

Stripped solvent is routed to a surge tank for reuse, while the weak aqueous product drains into a product concentrator surge tank. The aqueous waste from the extraction column, which contains essentially all of the impurities in the feed, drains to the batch quarantine tanks. Before release to secondary nitrate waste treatment, the aqueous waste sample is measured to verify that it meets uranium-limit specifications.

The product concentrator, controlled by the process computer, maintains a level setpoint in the surge tank and routes excess liquid through a condensate-heated preheater and a reboiler. Low pressure steam is used to percolate the uranyl nitrate solution through the reboiler heat exchanger. Vapors withdrawn containing the evaporated water are recondensed in an overhead condenser and reused for strip water in the solvent- extraction main process. Product density is automatically monitored and controlled by throttling the steam supply valve.

Laboratory tests confirm the quality of material accumulated in the accountability tanks. A computer calculates the uranium content in the accountability batch. The material is then transferred to storage for reuse in the fuel manufacturing operation.

## 4.3 Uranyl Nitrate Conversion

Following scrap recovery operations, the material recovered is uranyl nitrate. Uranyl nitrate conversion involves a precipitation process using ammonium hydroxide. (The flowchart for this process is shown in Figure 4.10.) Concentrated uranyl nitrate from the solvent extraction is transferred from scrap recovery product storage to uranyl nitrate storage. The uranyl nitrate storage system consists of six slab tanks. The configuration of the slab tanks ensures a favorable geometry to safeguard against inadvertent criticality.

Precipitation is accomplished by the controlled addition of ammonium hydroxide to the preheated uranyl nitrate in a mixing tank. Following completion of precipitation, the ammonium diuranate (ADU) slurry is transferred to a centrifuge feed tank. The feed tank mixes the ADU slurry with calciner-scrubber water and the clarifier underflow stream. The combined stream is then fed to a centrifuge.

The last processing step is conversion of the ADU to $UO_2$ via calcination. The $UO_2$ powder from the calciner is collected in containers, weighed, and forwarded to the $UO_2$ powder pre-treatment process.

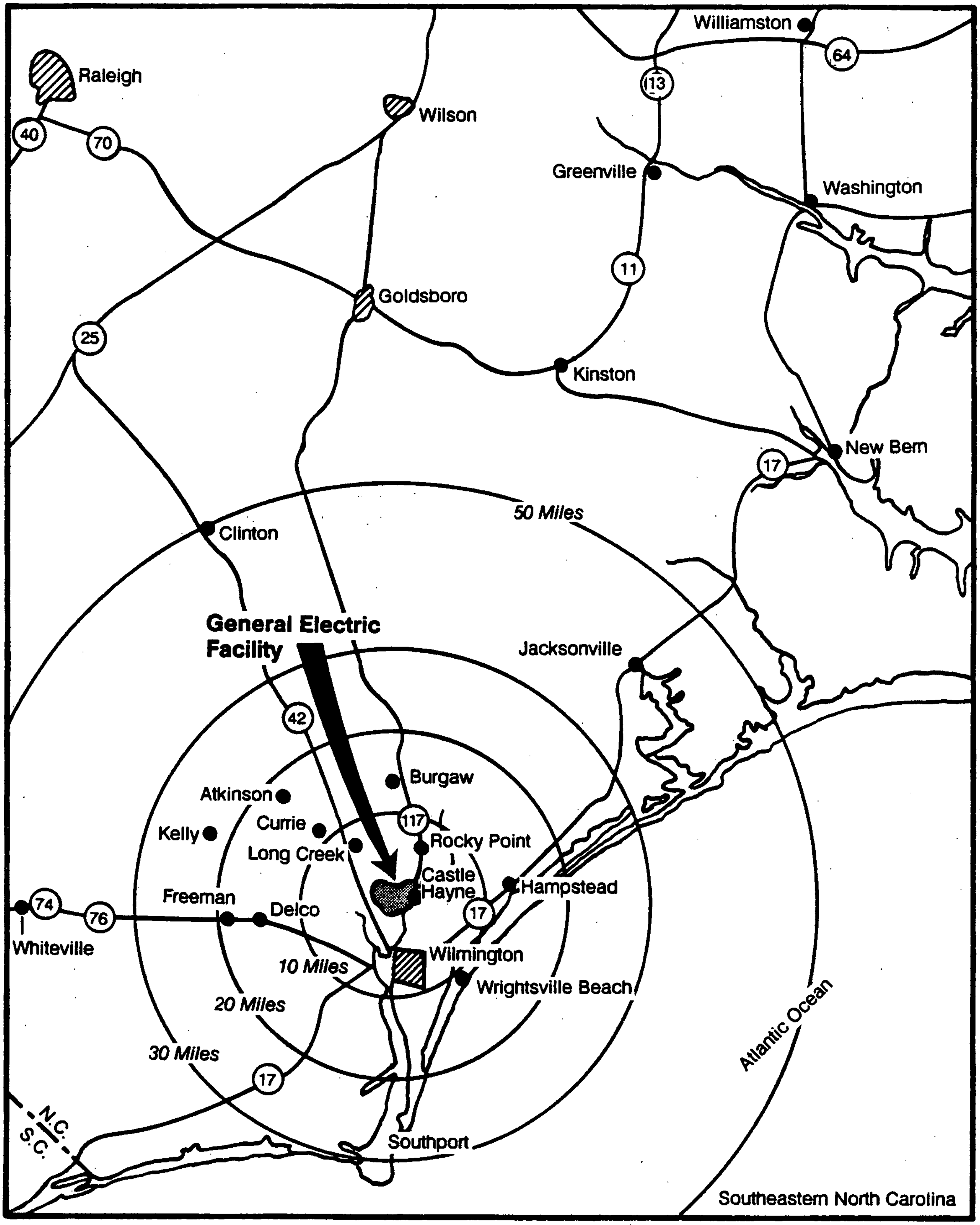


Figure 4.1 GE Nuclear Fuel and Component Manufacturing site location

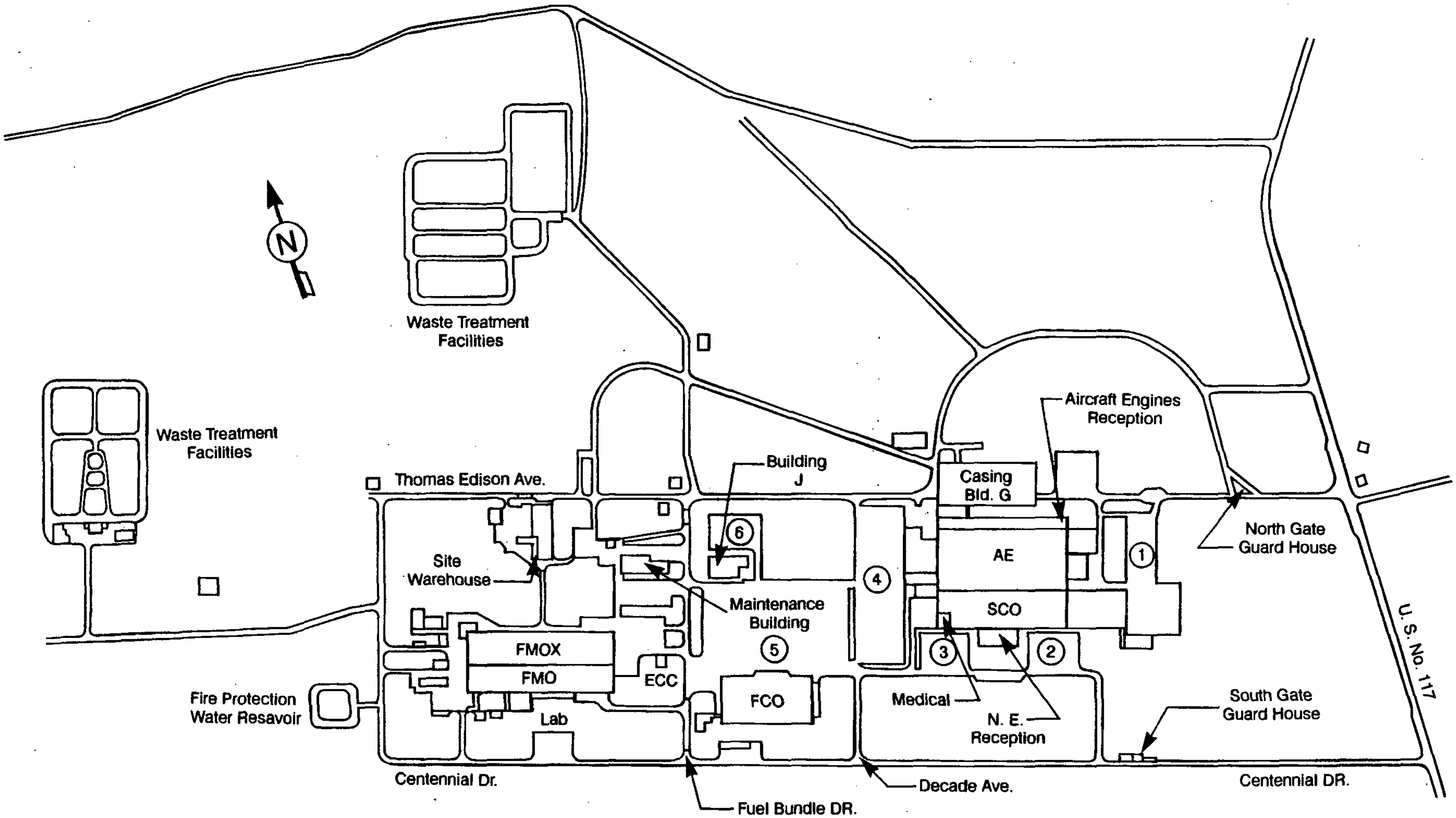


Figure 4.2 GE Nuclear Fuel and Component Operation: layout of major facilities

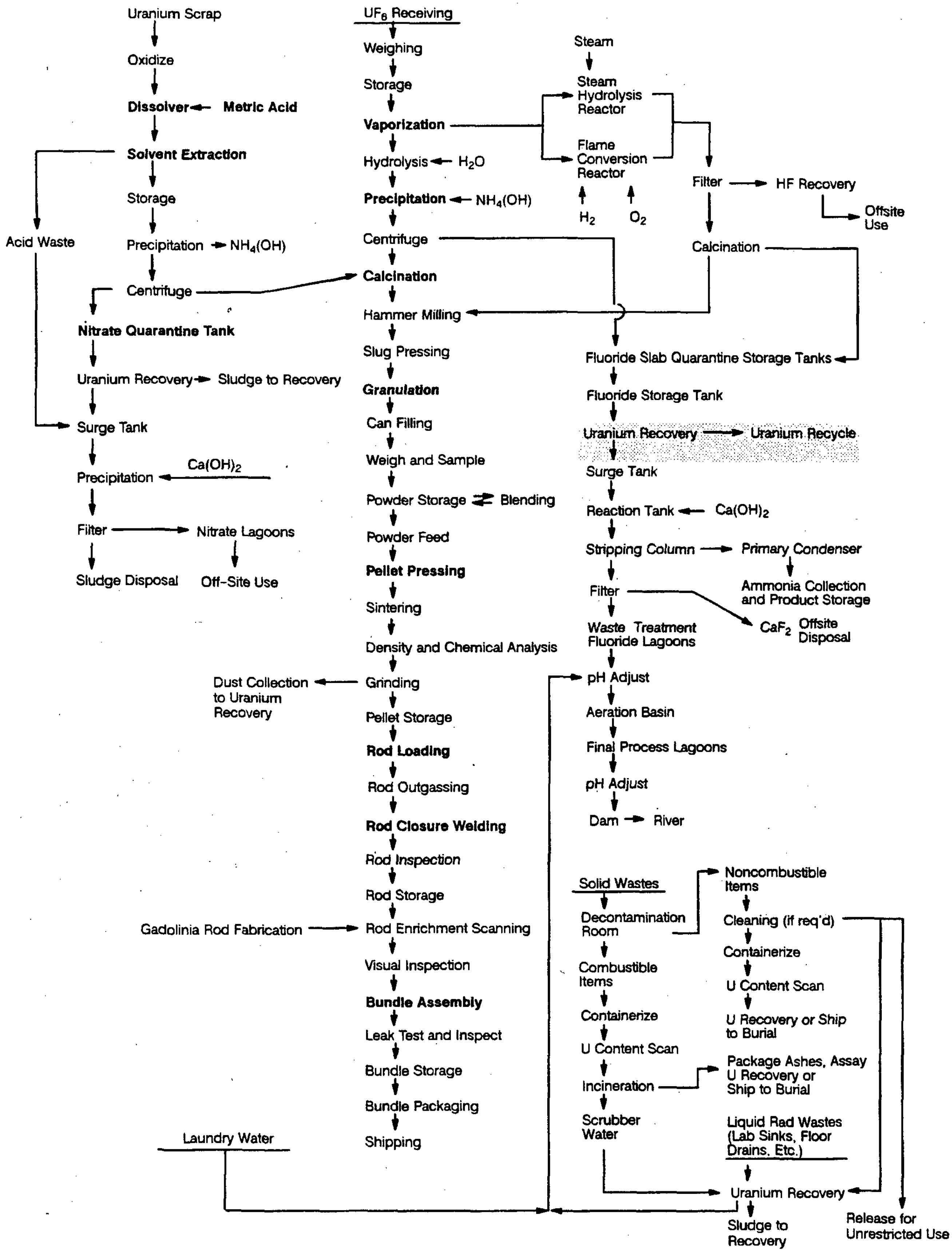


Figure 4.3 Uranium fuel process

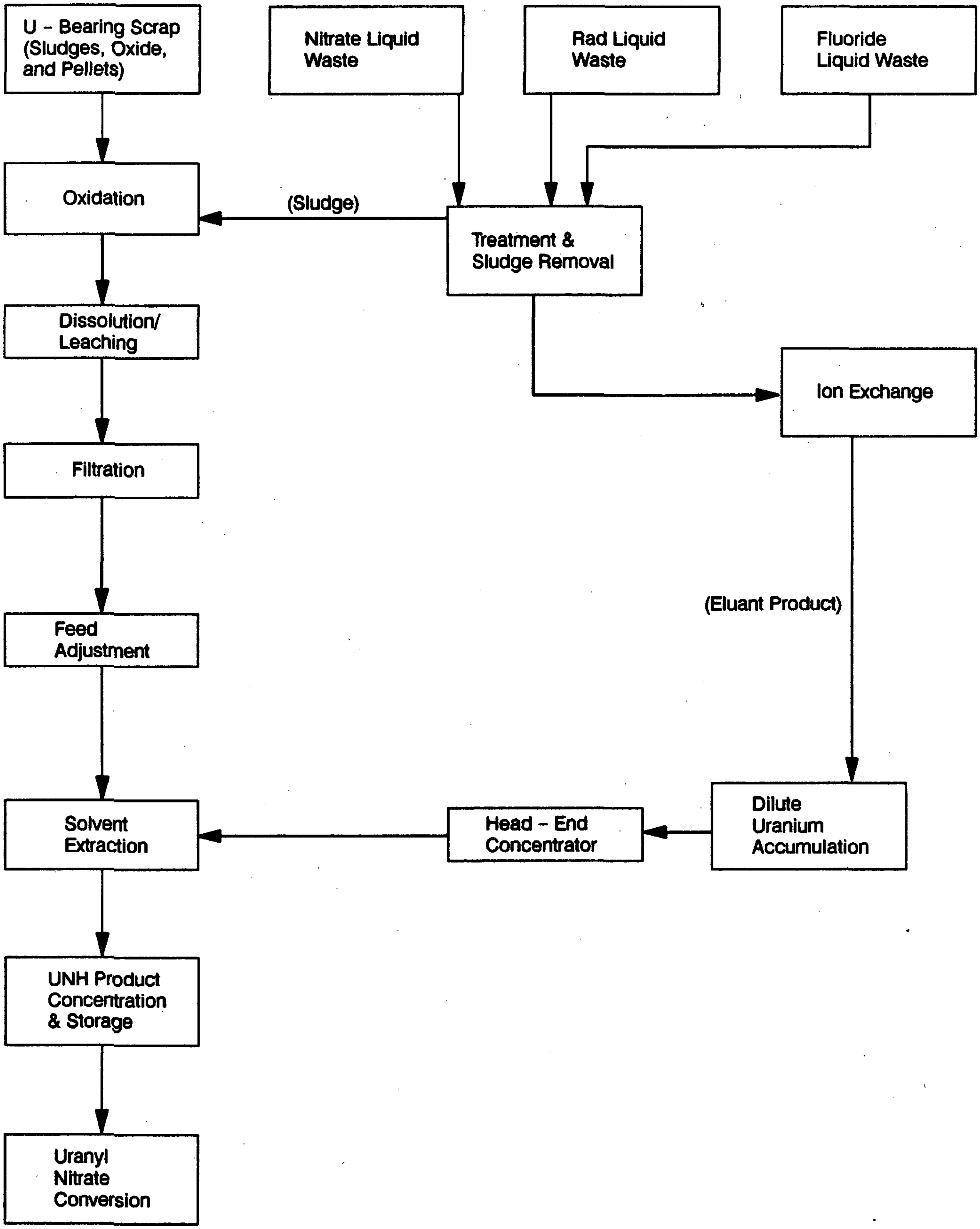


Figure 4.4 Uranium recovery and waste treatment operations

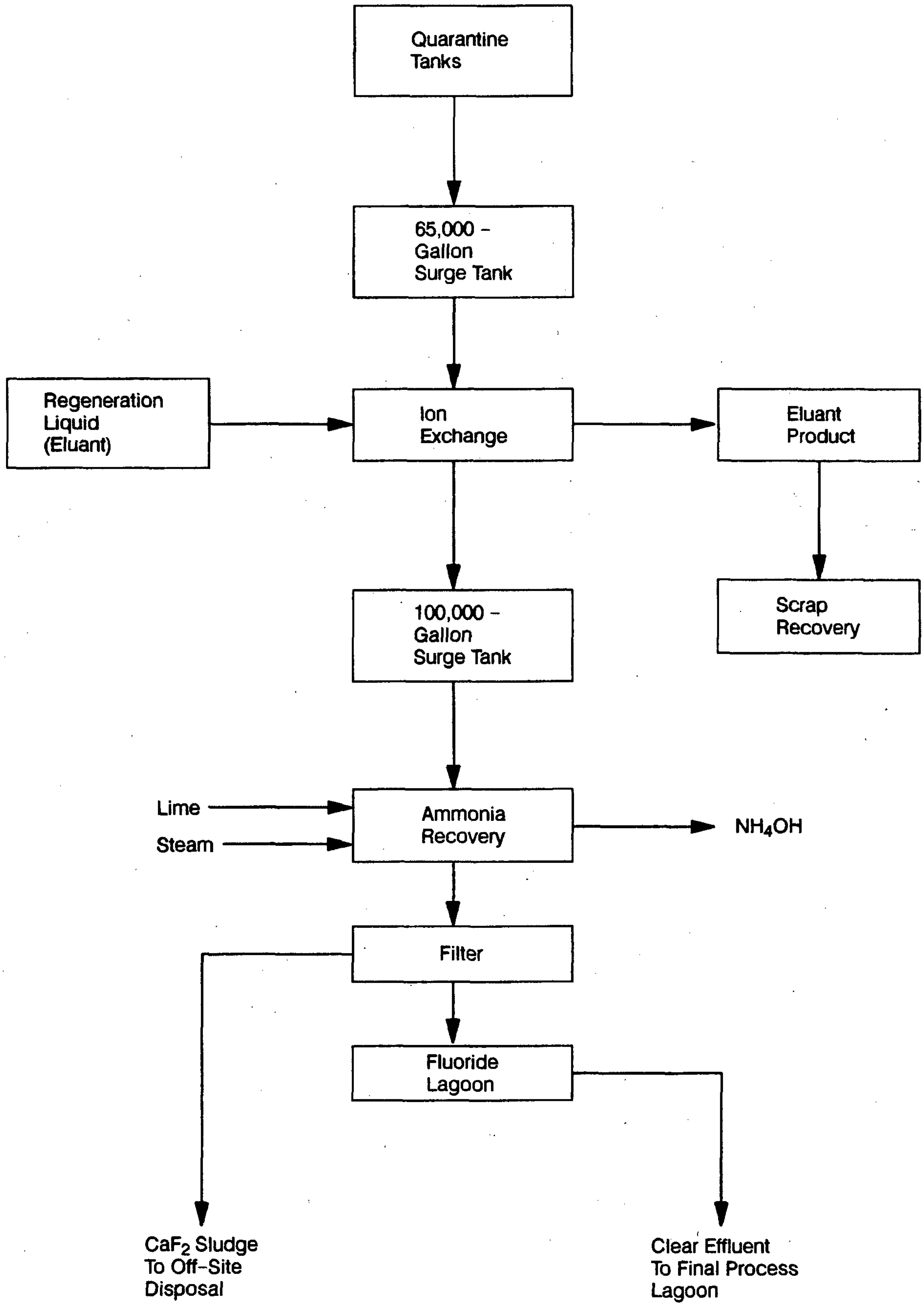


Figure 4.5 Fluoride waste treatment

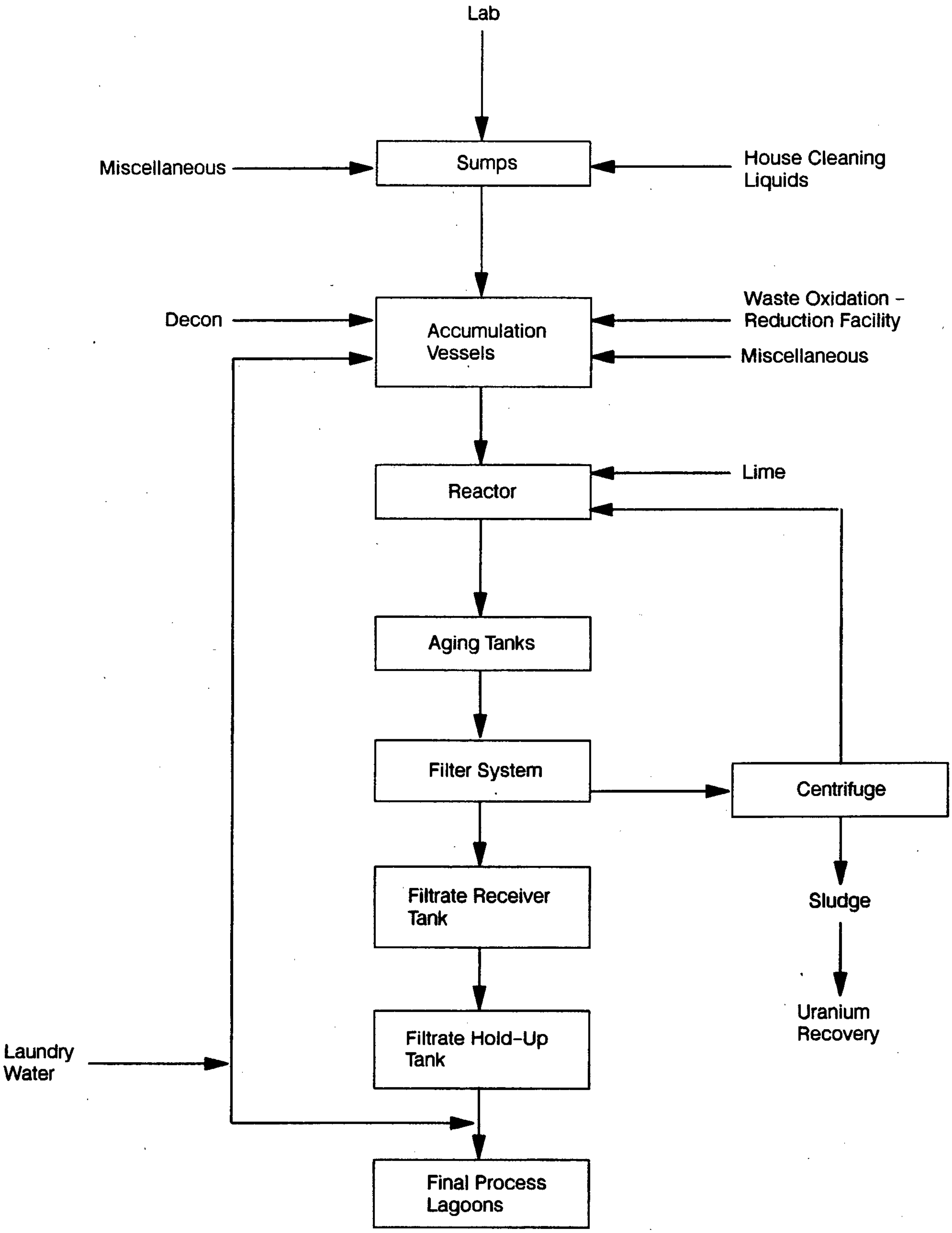


Figure 4.6 Radwaste treatment

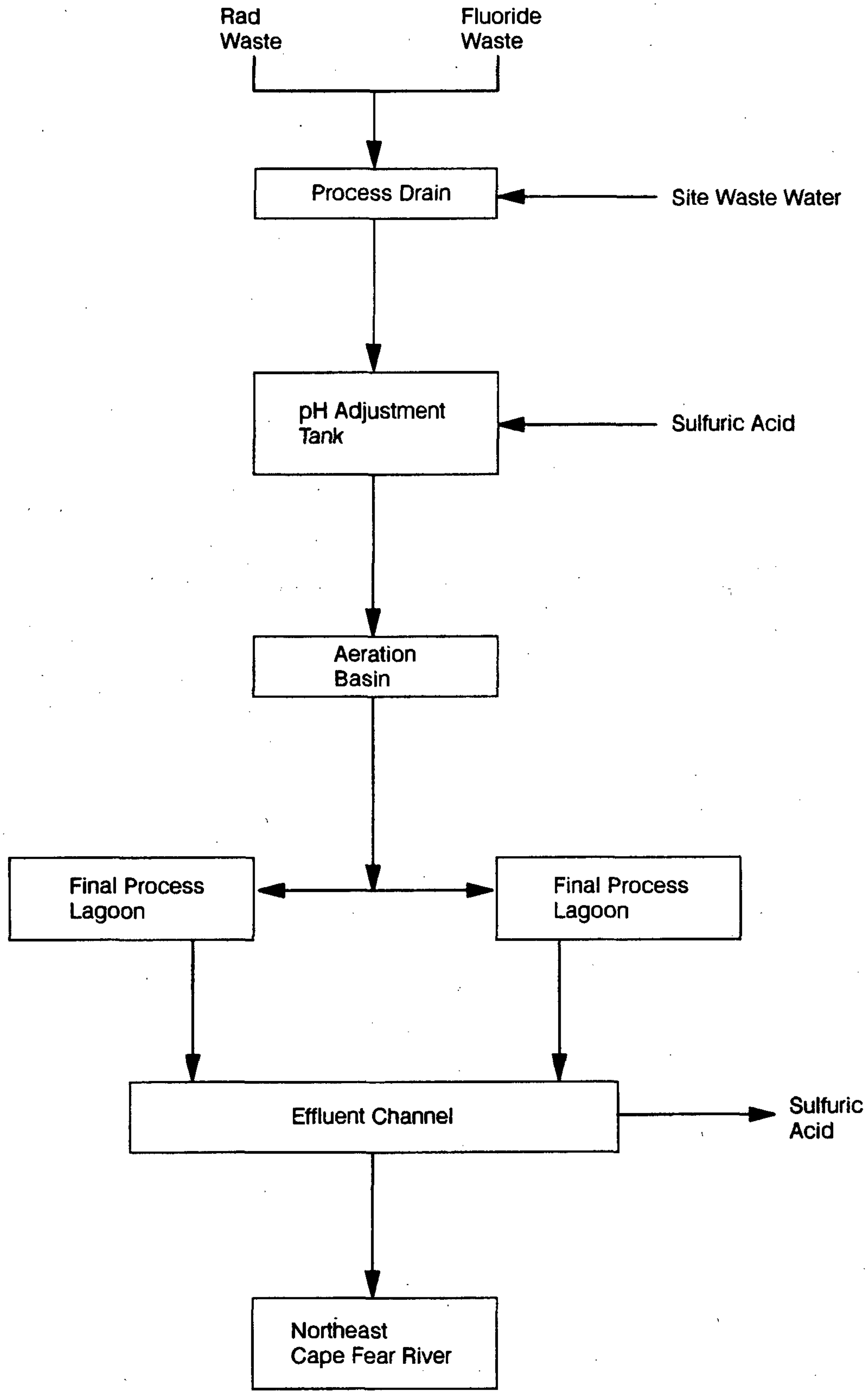


Figure 4.7 Final process lagoon system

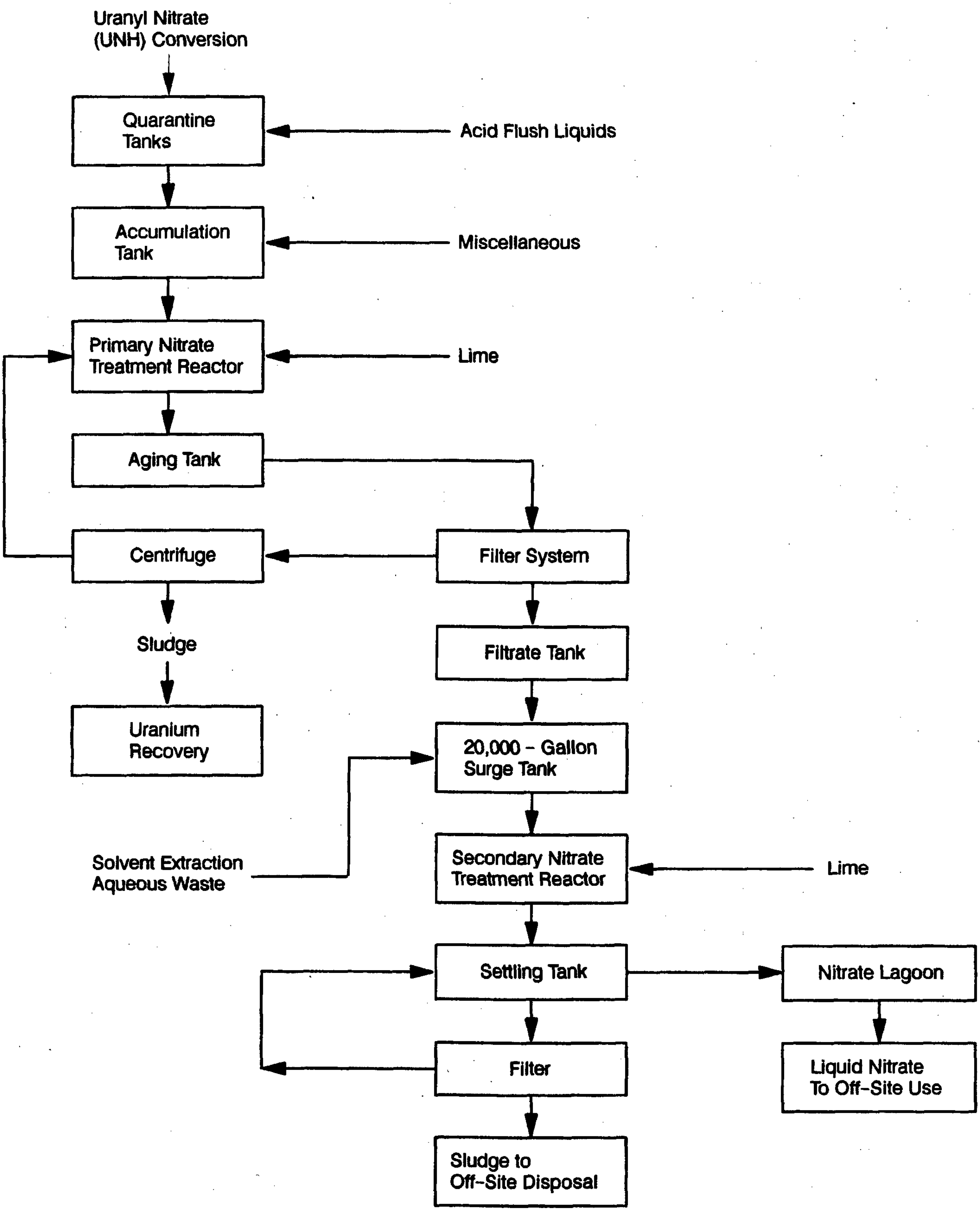


Figure 4.8 Nitrate waste treatment

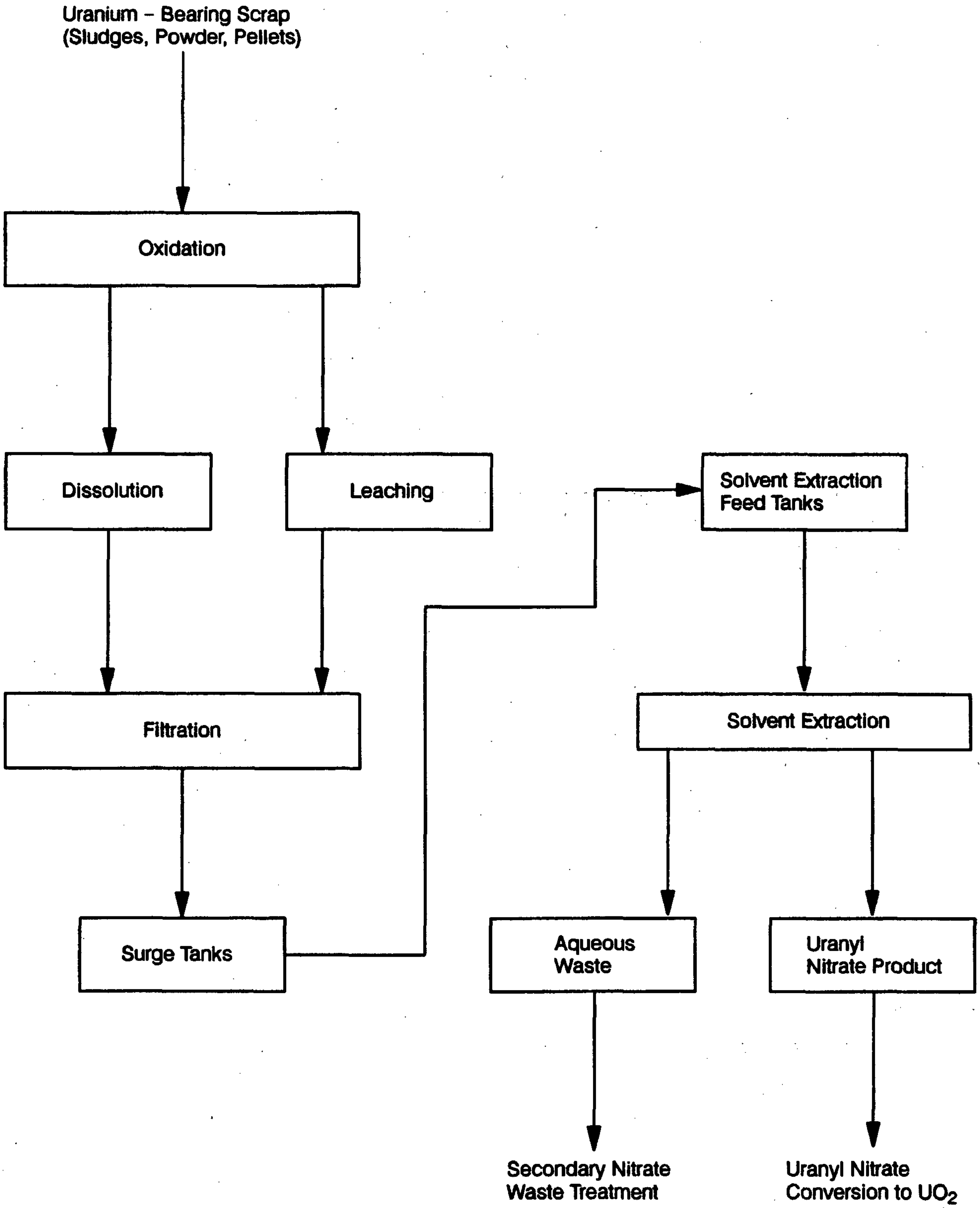


Figure 4.9 Typical scrap recovery process flow

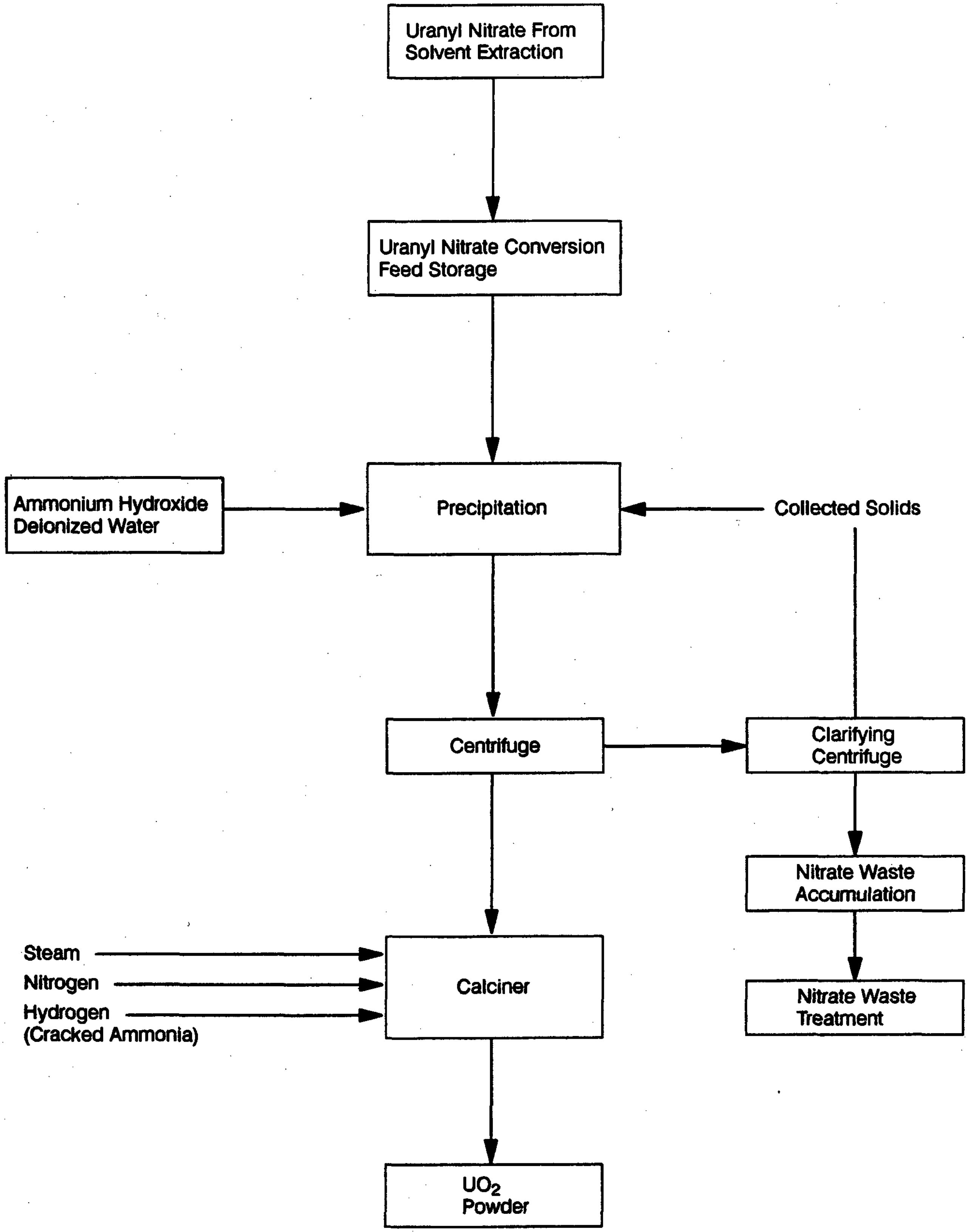


Figure 4.10 Uranyl nitrate conversion

# 5 EMERGENCY PLAN DESCRIPTION, RESPONSE, AND EVALUATION

This section describes and evaluates the emergency preparedness planning, response, and incident classification at the General Electric Nuclear Fuels and Components Manufacturing (NFCM) facility during the incident. This section also describes and evaluates the planning and response to the incident by State and local governments, and by the U.S. Nuclear Regulatory Commission (NRC). Table 5.1 provides a detailed sequence of events of the implementation of emergency preparedness plans and procedures and includes the licensee's basis for classifying the incident. Additionally, the team reviewed notifications made to the NRC, and to State and local governments for their accuracy, timeliness, and effectiveness.

## 5.1 General Emergency Planning and Preparedness Considerations

### 5.1.1 Emergency Planning Requirements

Emergency planning requirements for this facility are specified in Title 10, *Code of Federal Regulations*, Part 70.22(i)(1)(ii) and (i)(3), while requirements for criticality accident emergency procedures appear in 10 CFR Part 70.24(a)(3). Additional notification requirements are contained in 10 CFR Part 20.403 for incidents involving byproduct, source, or special nuclear material possessed by a licensee that may or could cause radiation exposures above specified limits. NRC's "Standard Format and Content for Emergency Plans for Fuel-Cycle and Materials Facilities" (NUREG-0720, November 1987), provides guidance as to what information should be included in a Radiological Contingency and Emergency Plan (RCEP). The agency's Regulatory Guide 3.42, Revision 1, September 1979, "Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70," describes acceptable methods for implementing the NRC's regulations, delineates techniques used to evaluate specific problems or postulated accidents, and provides guidance to license applicants. The RCEP is a license condition incorporated in its entirety by reference. The licensee may make changes to the plan as long as those changes do not degrade the effectiveness of the plan. Such changes need to be submitted to the NRC for review and approval within six months of the change. Changes which may degrade the effectiveness of the plan must be approved by the NRC prior to implementation.

### 5.1.2 The Licensee's Emergency Plan and Procedures

RCEP, Revision 7, December 1, 1988, was in effect during this incident. The licensee submitted this revision to the NRC for review and approval on January 19, 1989. NRC's evaluation determined that this change was acceptable and reviewers noted no planning deficiencies. Subsequently, NRC issued a Safety Evaluation Report (SER), March 20, 1989, stating that the plan change was acceptable. A revision to the licensee's RCEP of December 17, 1990, to address recent changes in the emergency planning requirements stated in 10 CFR Part 70, was being reviewed by NRC Office of Nuclear Material Safety and Safeguards (NMSS) and Region II as of the date of this report.

## 5.2 Detection and Assessment

Around 7:00 a.m. on May 29, 1991, sample measurements showed the uranium concentration in waste treatment tank V-104 to be 2333 ppm. This matter was discussed at a morning fuel manufacturing production meeting, where the licensee attributed the source of the problem to the solvent-extraction process upset which had occurred earlier that morning and the late evening of the preceding day. The licensee did not recognize this situation as an emergency condition, but handled it instead as a process upset, categorizing it at the Class II incident investigation level pursuant to their internal procedure. The licensee had precedents for handling this situation in this matter since they had experienced similar incidents involving the inadvertent transfer of high uranium-bearing solutions.

The licensee's initial actions involved ensuring that air sparging of tank V-104 continued so as to decrease the nuclear criticality potential that could be caused by the settling of the uranium in the bottom of the tank. Various levels of NFCM management were then apprised of the situation. None of the managers receiving this notification recognized the situation as an emergency or took exception with it being handled as a process problem. Licensee management did, however, assign a special response team to evaluate the situation and develop mitigation and remedial plans.

After evaluating the equipment, Waste Treatment Facility (WTF) design considerations, and resources available, the special response team recommended using the centrifuge located at the WTF to remove the uranium from tank V-104. After accepting this recommendation, WTF personnel made the centrifuge ready for service.

On the afternoon of May 29 about 3:00 p.m., the licensee placed a "courtesy call" to an NRC Senior Fuel Facility Inspector in Region II. The licensee termed it a "courtesy call" because they believed that the problem did not meet the criteria for formal reporting to the NRC. The NRC Senior Fuel Facility Inspector, not available at the time of the call, returned the call at about 3:45 p.m. and was advised of the problem. NRC then arranged for a more detailed briefing on this matter via a teleconference. During this briefing, NRC raised potential safety issues and emphasized the need to view the problem from an emergency-response perspective.

In summary, licensee management did not recognize the incident as an emergency, characterizing it instead as a process problem that met the criteria for a Class II internal investigation.

## 5.3 Emergency Classification

Chapter 3 of the RCEP addresses the licensee's emergency classification system. It presents a general description of events and conditions which may lead to the various emergency classifications, and includes a classification matrix that gives guidance for categorizing eight types of emergencies: bomb threat; confrontation, threat or theft; chemical spill/release; criticality; fire and explosion; radiological; severe weather; and transportation. A set of emergency implementing procedures have been developed for each of these eight emergencies. Although Procedure No. 5, "Criticality," stipulates actions to be taken during a criticality accident, it addresses emergency classification levels only to a limited degree. Under this procedure, only two classification levels are recognized: Unusual Event and Site Emergency. An Unusual Event is to be declared when a criticality alarm is sounding which proves to be false. A Site Emergency is to be declared when a criticality has occurred. Procedure No. 7, "Radiological," also provides some guidance on incident classification and recognizes three classes of emergencies: Unusual Event, Alert, and Site Emergency. Neither Procedure 5 nor 7 addresses a potential nuclear criticality accident. The guidance given in Procedures 5 and 7 is general and does not give examples of initiating conditions and numerical/quantitative action levels which could be used to aid in the classification process.

The situation confronting the licensee--a large quantity of uranium above critical mass limits in an unfavorable geometry tank with one criticality contingency remaining--was viewed by the NRC as having a high potential for a nuclear criticality accident. NRC believed that nuclear criticality safety had been significantly degraded because all criticality safety controls had been compromised, except for air sparging, which had to be actively maintained, i.e., a criticality could be a possibility with sparger failure. NRC further believed that the situation warranted an increased level of attention and response comparable to that afforded by activation of emergency planning provisions.

Licensee emergency plans and implementing procedures (referenced above) did not provide any guidance relative to the classification of potential nuclear criticality accidents. NRC regulations and guidance also do not currently address the classification of potential nuclear criticality accidents. One of NRC's roles in emergency response is to ensure that incidents are appropriately classified. For this situation, none of the licensee's emergency plan provisions or implementing procedures <u>clearly</u> lead one to a specific emergency classification. However, one section of the licensee's emergency plan, Subsection 3.1.2, contained classification guidance which closely matched this situation. Subsection 3.1.2 reads:

> <u>ALERT</u> This classification involves situations which could lead to identified hazard potentials. The situation has not yet caused damage to the facility nor harm to personnel and does not necessarily require an immediate change in facility operating status. Inherently, this is a situation in which time is available to take precautionary steps and/or mitigate consequences.

NRC made several requests that the licensee review the incident against the criteria in Subsection 3.1.2. Based on these requests, the licensee re-evaluated the situation and at 6:38 a.m. on May 30, 1991, declared an Alert emergency classification.

In summary, the RCEP and implementing procedures are incomplete or ambiguous with respect to classification guidance for this or a similar incident. The licensee's plan and procedures do not address potential nuclear criticalities. NRC regulations and Regulatory Guides are also deficient with respect to classification requirements and guidance for potential nuclear criticalities.

## 5.4 Offsite Notification

### 5.4.1 NRC Notification

At 3:00 p.m. on May 29, 1991, the licensee initiated a "courtesy call" to the NRC Region II Fuel Facility Inspector (8 hours following the identification of the incident), as discussed in Section 5.2 of this report. The NRC Senior Fuel Facility inspector returned the call at 3:45 p.m., and the licensee provided the information listed in Table 5.1. Around 5:00 p.m. (10 hours following the identification of the incident), the NRC Headquarter's Operations Officer (HOO) was contacted by an NRC headquarter's staff person concerning the incident and NRC's entry into the Standby mode. Shortly thereafter, the Operations Center was contacted by the Deputy Director, Division of Reactor Safety and Safeguards, Region II, who requested that a teleconference bridge via the Emergency Notification System (ENS) be established between the NRC and the licensee at 6:30 p.m., briefed the HOO concerning the incident, and informed the HOO that the agency (NRC) was in standby. The licensee declared an Alert at 6:38 a.m. on May 30, 1991 and made its official notification to the NRC of the incident at 6:40 a.m. on May 30, 1991 (23 hours and 40 minutes after the incident was first identified). The NRC was notified of the termination of the Alert at 3:40 a.m. June 1, 1990 (approximately 45 hours after the incident was first identified).

### 5.4.2 State and Local Government Notification

The State of North Carolina radiological health authorities were first advised of the incident by the NRC Region II office on May 29, 1991, at approximately 6:30 p.m. The State was concerned that it had not been notified initially by the licensee and that the NRC had not notified the State at 3:45 p.m., when the incident was first identified to the NRC. The licensee first discussed the incident with State radiological health officials around 9:30 p.m. on May 29, 1991, when a licensee nuclear safety engineer provided a status update to the State's radiological emergency planner. When the licensee declared the Alert at 6:38 a.m on May 30, 1991, notifications were made to the State in accordance with the RCEP. Other discussions with the State are shown in Table 5.1.

New Hanover County, in which the facility is located, received its initial notification and information about the incident from the State shortly after NRC informed the State. The county was informed of the Alert declaration and termination at the times stated in Table 5.1. Although these notifications were made to New Hanover County Officials, the Director of the New Hanover County Department of Emergency Management expressed concern about the lack of formal updates, guidance, or recommendations from the licensee regarding appropriate response actions. The lack of information hindered the New Hanover County officials in responding to public and media inquiries.

## 5.5 The Role of State and Local Government

State and local governmental agencies did not implement their respective emergency response plans or activate their emergency response facilities as a result of this incident. The State of North Carolina radiological protection authorities monitored licensee incident response activities via the ENS telecommunication bridge, but State personnel did not deploy to the site. The State maintained an increased state of readiness to deploy and continued to monitor the information being transmitted over the teleconference bridge.

New Hanover County emergency representatives did not deploy to the site either. The county received initial and final notifications regarding the emergency from the licensee and information on the response effort from North Carolina radiological protection officials. Although the County emergency facilities were not activated, the Director of the Department of Emergency Management handled numerous requests for information from the public and the media.

The requirements in 10 CFR 70.22 specify that "The licensee shall invite offsite response organizations to participate in the biennial exercise. Participation . . . although recommended is not required." Based upon interviews with the licensee and State and local government personnel, the team determined that the State and local government response organizations had been invited to participate in drills and exercises. However, the State and local government departments with responsibility for emergency response oversight have not actively participated in drills or exercises, although, local fire departments and law enforcement agencies have. The team noted that State and local government participation similar to "partial participation" activities at nuclear power plant exercises was not an option considered.

In summary, the team finds that emergency plans are lacking with respect to the flow of information to State and local governments, to arrangements for an offsite presence at the licensee's Emergency Control Center (ECC), and for the licensee's presence at the State and County Emergency Operations Center (EOC). There is also a lack of offsite planning and coordination with respect to public notification and information between the licensee and State and local governments. The State expressed concern to the NRC about the need for prompt reporting of events that could lead to a nuclear criticality accident and that information concerning incidents at these types of facilities be provided in an expedited and structured manner.

## 5.6 Public Information

Following discussions with NRC on the evening of May 29, 1991, the licensee agreed to issue a press release concerning the incident. On the morning of May 30, 1990, they provided a draft copy of the press release to the NRC that characterized the incident as a chemical emergency. The NRC felt that the licensee's press release did not accurately characterize the situation; NRC, therefore, issued a separate press release that discussed the incident from a nuclear criticality safety perspective. The team concludes that the licensee's press release was misleading and not responsive to the public's need.

The State of North Carolina radiological protection authorities expressed concern to NRC about the licensee's news release. The State further expressed the need for a more coordinated approach to developing and issuing press releases involving the licensee and appropriate government response agencies.

The licensee appointed a Public Information Officer (PIO) to handle media and public inquiries, which was done at the PIO's normal duty station - the Human Resources Offices - rather than at the licensee's ECC. Even so, this did not hinder the PIO's activities.

On May 30, 1991, the NRC dispatched the NRC Region II Public Affairs Officer (PAO) to the site to handle media inquires for the NRC. Information provided to the media was coordinated with the NRC site team leader and with NRC headquarters and the NRC Region II office. Based on numerous requests for information from local and regional media, the NRC held a press conference on the afternoon of May 31, 1991. The licensee declined to participate in the press conference.

The New Hanover County Department of Emergency Management received numerous requests for information from the media and general public. County officials said that they experienced some difficulty in answering these requests because of the lack of information from the site.

In summary, the team finds that NRC's guidance and the licensee's RCEP emergency planning provisions are lacking with respect to the dissemination of information to the general public and to news media.

## 5.7 NRC and Licensee Interactions

Following the licensee's initial notification, an open ENS teleconferencing link was established between the licensee and NRC (NRC headquarter's operations office and regional base team). The open ENS was used extensively for the exchange of technical information and status briefings. Based on information provided during the courtesy call and on subsequent NRC management review, an NRC site team was dispatched to the site.

During the evening of May 29, the licensee provided information and briefings via the ENS teleconference link. In these discussions, the NRC emphasized the safety significance of the problem and encouraged the licensee to consider criticality control contingency measures, increased staffing to cope with the problem, and emergency response actions. As a result of prompting by the NRC, the licensee augmented its staff with operations, maintenance, engineering, management, and safety personnel and made arrangements for 24-hour coverage. The NRC questioned the licensee in detail concerning contingency measures to keep the air sparge operating and options available in the event of air sparger failure. Following these discussions, the licensee provided the NRC with a status report on the availability of air supply systems at the fuel manufacturing plant, identified backup air compressor in the WTF area, made arrangements for a portable air compressor to be brought to the WTF, and devised and field-fabricated a backup sparger to augment the built-in sparger in tank V-104. As a backup measure to cope with possible air sparger failure, an insoluble poison, a form of gadolinium oxide, was obtained and placed adjacent to tank V-104. Later discussions between the licensee and NRC resulted in the licensee obtaining a soluble poison, boric acid, for use in mitigating the consequences of a sparging failure.

From the onset of the problem (the determination of high uranium concentrations in tank V-104), the licensee treated the incident as a process upset. During the morning of May 30, at the urging of the NRC, the licensee re-evaluated its emergency plan requirements and decided to classify the incident as an Alert emergency classification (see Subsection 5.3 above).

As shown in Table 5.1, the licensee provided information concerning samples taken from various tanks. NRC was concerned that there were possible errors in the data because of the sampling methodology being used by the licensee in obtaining them. As a result, the licensee re-evaluated its sampling methodology and developed a sampling plan that ultimately provided reproducible and reliable results.

An NRC site team was dispatched between 5:15 p.m. and 5:30 p.m. on May 29, 1991, and arrived at 12:55 a.m. on May 30, 1991. The team was briefed and commenced its independent assessment and monitoring of the situation by walkdowns and physical inspections. The team inspected the air compressor system, portable back up compressor, temporary sparging lines, and observed the tank sparging. The results of these activities and requests for additional information, as well as a confirmation of actions taken by the licensee, were given to NRC headquarters and Region II during briefings by the site team leader, as shown in Table 5.1.

During discussions with the licensee concerning mitigation strategies, the NRC suggested methods the licensee should consider in reducing further the amount of uranium in tank V-104 to thereby further reduce the potential for a nuclear criticality accident. Based on these discussions, the licensee began evaluating options to reduce the critical mass by transferring the contents of tank V-104 to other tanks. Two on-site tanks, V-109A and V-109B located adjacent to tank V-104, were identified and made available to receive a portion of V-104's contents. During the late evening of May 31, 1991, the licensee advised the NRC that they planned to reduce the critical mass by partial transfer to tank V-109B. The NRC concurred with the plan. See Section 3 of this report for further details of this operation. During the early morning of June 1, 1991, this partial transfer was completed. With the quantity of uranium below the critical mass limits, the Alert was terminated at 3:20 a.m. on June 1, 1991.

Several licensee personnel interviewed by the IIT stated that this incident gave them their first experience in interacting with the NRC incident response organization and that they were overwhelmed at the intensity with which the NRC pursued the facts and circumstances concerning the incident. The licensee did not fully comprehend NRC's role in its response to incidents, nor did it understand the concept of operations and functions of NRC's Incident Response Organizations: Headquarters Operations Center, Region Base Team, and Site Team. When the NRC site team arrived during the incident, the licensee was under the initial impression that the team was there for an inspection.

## 5.8 NRC Response to the Incident

The team determined that the NRC was able to mobilize its resources and respond to this incident in accordance with the agency's incident response plan. As discussed in Section 5.7, the NRC interaction with the licensee contributed to an effective resolution of the incident. There were, however, apparent implementation problems with the NRC's response. NRC Headquarters and regional response plans for fuel facilities have not been formally coordinated. This led to loosely structured interactions with the licensee. The team found that on occasion NRC's questions and suggestions were not well coordinated. NRC staff at headquarters and Region II made suggestions to the licensee in some instances without first evaluating the implications of the suggestions. They also posed questions to the licensee before fully coordinating the intent of the questions among themselves. For example, the licensee made surface radiation readings at tank V-104 with hand-held radiation monitors in response to NRC questions and suggestions regarding assurance that the uranium was not settling in the bottom of the tank. Since the incident involved a potential criticality in the tank, this action placed the persons conducting the monitoring at potential risk. More structured deliberation on the part of NRC would have led to a fuller realization of the benefit of having such data versus the potential risk to personnel in how it was obtained. The team believes that this lack of preparation and coordination resulted from a lack of planning specific to a potential criticality incident.

## 5.9 Findings and Conclusions

- The licensee did not consider the incident serious enough to be classified as an emergency.

- The licensee's RCEP and Implementing Procedures are ambiguous, incomplete, and do not consider a potential criticality. NRC requirements and guidance as currently written do not require licensees to assess incidents based on loss of nuclear criticality safety barriers or a potential nuclear criticality.

- Once the Alert was declared, the licensee made appropriate notifications in accordance with the RCEP.

- The use of the ENS telecommunications system greatly enhanced the flow of information between the NRC and the licensee.

- The team finds that NRC's guidance and the licensee's RCEP emergency planning provisions are lacking with respect to the dissemination of information to the general public and to news media.

- The licensee response organization tended to evaluate contingency measures thoroughly; however, the response organization was not always proactive in this effort and, in some cases, had to be urged by the NRC to take action.

- The licensee did not fully understand the NRC's role in responding to the incident.

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 5/28/91 | 9:30 p.m. | Process control operators in the URU control room detected problems in controlling the interface of the solvent-extraction process. |
| 5/29/91 | 5:20 a.m. | The licensee shut the solvent-extraction process down after a measured sample from tank V-291 indicated 6977 ppm of uranium; prior to this measurement, four unmeasured tanks had been transferred to tank V-103. |
| | 6:30 a.m.- 7:00 a.m. | GE NFCM management was notified that a measured sample from tank V-104 in the Waste Treatment Facility (WTF) indicated 2333 ppm of uranium. |
| | 10:45 a.m. | The Manager, Regulatory Compliance, briefed the acting manager of NFCM, who decided to give increased attention to the response. |
| | 12:00 noon | A special response (Technical Evaluation) team met to discuss safe recovery and control contingency; they then briefed the Manager, Regulatory Compliance. |
| | 3:00 p.m. | The licensee initiated a "Courtesy Notification" to NRC's, Region II. |
| | 3:45 p.m. | NRC RII returned the call and was briefed by the licensee as follows:<br>• The SX system appears to be the source of the uranium.<br>• Quantities of uranium transferred were estimated at 140 and 150 kgs.<br>• A single nuclear criticality safety contingency was operating effectively.<br>• A plan was formulated to safely recover the material (by centrifuge).<br>• The sample density after three hours of settling was approximately 1.15 g/cc.<br>• One centrifuge was available (another is out of service and will not be available).<br>• The "heel" in tank V-103 was less than critical mass. |

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 5/29/91 | 3:45 p.m. | • The site was not in a site emergency status but a Class II internal incident status.<br>• The estimated minimum total uranium for criticality is 50 kgs.<br>• The estimated time of removal to get below critical mass is early a.m. 5/30; estimated time for complete removal was approximately noon 5/30.<br>• The tank is at a remote part of the site; the centrifuge is below it; there are approximately four personnel in this area.<br>• The criticality detector is located about 100 feet from the tank.<br><br>During this teleconference, NRC suggested that the licensee:<br><br>• Consider activating the Emergency Plan because only a single barrier (air sparging) initiated against criticality.<br>• Acquire poisoning materials should sparging fail.<br>• Ensure that measurements are not influenced by organic material in solution.<br>• Keep personnel active, until it's clear that criticality is no longer possible.<br>• Issue a press release instead of, or at least before, the NRC. |
| | 5:44 p.m. | NRC went into Standby mode. |
| | 6:30 p.m. | Region II notified the State of North Carolina.<br><br>NRC initiated an ENS teleconference bridge.<br><br>The licensee briefed NRC Headquarters and Region II regarding:<br>• Technical information exchange.<br>• Recovery, staffing, and communication planning. |

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 5/29/91 | 6:30 p.m. | NRC noted that:<br>● The NRC Site Team was dispatched, gave its estimated arrival time, and requested that the licensee quarantine the SX process and appropriate equipment. |
| | 8:45 p.m. | Commissioner Assistant's briefing. |
| | 8:55 p.m. | Licensee operations and maintenance activities included locating a portable air compressor for the WTF as a back-up for air sparging and switching the HVAC scrubber to radiation waste to minimize the volume in tank V-103. |
| | 9:15 p.m. | NRC completed notifications of other regions and Federal agencies. |
| | 11:00 p.m. | The licensee shut down most chemical conversion operations and evacuated personnel from the Uranium Recovery Lagoon area. |
| | 11:30 p.m. | Teleconference held between NRC and the licensee.<br>● NRC decided to maintain its Standby mode.<br>● The licensee agreed to issue a press release at approximately 7:30 a.m., 5/30; NRC would not issue one before that.<br>● NRC asked the following questions:<br>Is 50 kgs really the minimum needed for critical mass?<br>Why does it take so long to analyze the contents of recovered sludge?<br>How reliable is the air sparging system?<br>Is monitoring adequate to protect the workers? |
| 5/30/91 | 12:55 a.m. | The NRC Region II Site Team arrived on site. |
| | 1:15 a.m. | The licensee briefed the NRC Site Team on incident status. |
| | 4:00 a.m. | The NRC Site Team briefed headquarter's and region-based team on independent assessments. |

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 5/30/91 | 6:38 a.m. | The licensee reevaluated its emergency plan and procedures based on questions from NRC. They determined that Site Status should be classified as an Alert; they implemented Procedure 7, "Radiological" of RCEP. State and local governments were notified; the licensee noted that a press release would be issued shortly. |
| | 6:40 a.m. | The licensee notified NRC that an Alert was declared. |
| | 7:17 a.m. | In a teleconference, the licensee reviewed its response to the event to date; total material removed was 83 kgs (sludge). |
| | 8:00 a.m. | The licensee issued a news release; NRC held a management briefing. The licensee reported sample results for tanks V-103, V-104, and the uranium enrichment in tank V-104. |
| | 1:00 p.m. | The licensee reported to NRC on sampling at three different levels within tank V-104. |
| | 2:30 p.m. - 3:45 p.m. | The State of North Carolina discussed concerns with the licensee's press release. |
| | 4:00 p.m. - | NRC conference call. |
| | 5:00 p.m. | Licensee briefing:<br>• 240 kgs sludge were removed; no update on uranium fraction.<br>• Contact radiation readings on the side of tank V-104 are 0.05 - 0.10 mR/hr and at discharge pipe.<br>• Boric acid was now available.<br>NRC asked if the addition of boric acid will redissolve significant amounts of uranium or otherwise affect the process chemistry.<br>• No other areas of the plant are affected. |

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 5/30/91 | 5:00 p.m. | • The licensee proposed a less conservative but faster removal when remaining uranium reaches 100 kgs. NRC asked the licensee to quantify the benefit from the faster process to show that it outweighs the increased risk. |
| | 6:30 p.m. | Region II briefed the State of North Carolina; the State was invited to join the next teleconference call. |
| | 7:30 p.m. | NRC asked whether using 100 kgs as the criterion for implementing the less conservative (auto-sparging) procedure leaves a reactivity margin of only 0.97. NRC requires 0.90 and asked the licensee to translate that to kgs remaining in the tank. |
| | 11:00 p.m. | Licensee briefing:<br>• Total sludge removed 270 kgs; uranium fraction still uncertain.<br>• Rad contact readings at bottom of tank indicate that uranium is not accumulating at the bottom.<br>• In response to the addition of boric acid-4.6 g/l settling was slower and uranium concentration in solution was higher.<br>• The licensee proposed and NRC agreed only to routine sampling and that the licensee will continue current procedures and forget about earlier changes when the remaining uranium reaches 100 kgs. |
| 5/31/91 | 12:10 a.m. | Licensee briefing:<br>• Total sludge removed 285 kgs.<br>• Clutch problem repaired and centrifuge being returned to service; sparge was continued while centrifuge was secured.<br>• Tank chemistry as of 10:00 p.m. shows 172 ppm at top, 943 at middle, and 1112 ppm at bottom; three cans of sludge were sampled; the uranium concentration is 15 percent. |

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 5/31/91 | 1:25 a.m. | The licensee reports a centrifuge problem; contact readings at 12:30 a.m. were 0.15 mR/hr. |
| | 2:25 a.m. | Region II was informed that the licensee has centrifuging back in service as of 1:55 a.m. |
| | 3:00 a.m. | The licensee initiated an improved quality measurement plan to better quantify tank inventory. |
| | 2:00 p.m. | A conference call was held with NRC (Headquarters, Region II, Site Team), the licensee, and the State of North Carolina for an incident update. |
| | 8:00 p.m. | The licensee received a Confirmation of Action Letter from NRC. |
| | 9:00 p.m. | A conference call was held (among the same participants as for the 2:00 p.m. call) for an incident update. |
| | 10:30 p.m. | The licensee began to transfer 7,000 gallons from tank V-104 to tanks V-109A and V-109B at WTF. |
| 6/1/91 | 12:01 a.m. | A conference call was held with the NRC Site Team, Region II, and HQ Operations Center (ET). |
| | 12:45 a.m. | A conference call was held between NRC management and the licensee. |
| | 1:15 a.m. | A conference call was held among cognizant NRC managers. |
| | 3:15 a.m. | The licensee completed the transfer of 7,490 gallons to tanks V-109A and V-109B to provide less-than-critical mass in tanks V-104, V-109A, and V-109B. |

Table 5.1 Incident response chronology

| Date | Time | Description of Incident |
|---|---|---|
| 6/1/91 | 3:20 a.m. | The licensee terminated the Alert status following discussions with NRC and the State; The Emergency Notifications System (ENS) was disconnected (NRC went out of Standby mode); uranium recovery-centrifuge operations continue. |
| | 8:00 a.m. | Commissioner's Assistant briefing on the incident. |
| | 1:10 p.m. | An NRC conference call concerning sludge transfer operations |
| | 8:25 p.m. | An NRC conference call concerning sludge transfer operations |
| 6/2/91 | | The licensee continued uranium recovery (centrifuge) operations. |
| 6/3/91 | ~10:30 a.m. | The licensee completed uranium recovery (centrifuge) operations, having removed approximately 136 kgs of uranium. |

# 6 HUMAN FACTORS CONSIDERATIONS

This section describes and evaluates personnel and operator-interface system performance during the incident which led to a potential criticality safety problem in a waste treatment tank in the Uranium Recycle Unit (URU) at the General Electric Company's Nuclear Fuel and Component Manufacturing (NFCM) facility near Wilmington, North Carolina. Certain features of the facility and its operations contributed to the potential seriousness of the incident:

- URU operator interface system
- Operating procedures
- URU operator training
- Supervisory and management oversight

An evaluation of personnel performance, procedures, and management oversight of the licensee's emergency plan and response to the incident appears in Section 5 of this report. A similar evaluation of maintenance personnel is provided in Section 9. Personnel and systems performance during uranium recovery operations are described in Sections 3, 5, and Appendix B.

## 6.1 URU Operator Interface System

### 6.1.1 System Design Description and Staffing

The URU is composed of several systems that are used in four processes to remove uranium from waste products from the nuclear fuel manufacturing process: solvent extraction, nitrate waste treatment, fluoride waste treatment, and radwaste treatment. There are four rotating shifts of operators at the URU. The normal shift compliment consists of two A operators, one assigned to the control room and one assigned to the operating floor; four B operators, one assigned to the control room and three assigned to the operating floor; one C operator assigned to the operating floor; and one A and one B operator assigned to the waste treatment facility. Normally, each shift works eight hours except when there is a need to cover for someone's absence. Operators average less than 1 hour of overtime per week. A operators have overall responsibility for the area they are assigned to, i.e., the control room, the operating floor, or the Waste Treatment Facility. B operators receive instructions from their respective A operators. In the control room, the A operator supervises URU operations and monitors the SX process. The B control room operator monitors the radwaste, nitrate waste and fluoride waste processes. The A floor and waste treatment operators coordinate process activities and provide the B operators with instructions and guidance. The operating floor of the URU is physically separated from the control room by contamination control barriers. The Waste Treatment Facility is approximately one-fourth mile away from the URU. Communications between the control room and the floor are maintained by paging and telephone systems.

Supervisory oversight for URU operations is maintained by a Chemical Area Coordinator. In addition to the URU, the area coordinator is the supervisor for operations in the $UF_6$ conversion area and, during non-day time and weekend hours, the fuel support area, the pellet production furnace and the gadolinia furnace. Section 6.4 discusses supervisory oversight in more detail.

The URU process control system is a commercially available digital control system which employs an electronic console unit for operator interface. In the URU control room, there are two operator consoles, each with two-color cathode ray tubes (CRT) and a printer (Figure 6.1). The consoles provide real-time interactive graphics displays of all process parameters and provides the operators with the means to monitor and control complex batch and sequential process operations through the digital control system. The control system automates many tasks that were traditionally performed manually by an operator and that required constant attention to maintain control. The automated system is, by contrast, capable of initiating each phase of a process, monitoring process variables, and cueing discrete activities based on time or process conditions. The intent of the automated control system design was to improve production efficiency and minimize the chance of an operator error. The operator essentially monitors overall operations and controls process functions instead of individual components or systems. The operator monitors the process by scanning the various displays to assure that the system is functioning properly. The system also provides for manual adjustment or TUNE mode capability for certain process and control loop parameters. The TUNE mode also allows operators to temporarily bypass or override certain automatic functions. For example, using the TUNE mode to transfer the contents of one tank to another, the operator would specify at the control console by tank numbers which tank was to be emptied and which was to be filled rather than specifying that certain valves and pumps be activated individually to perform the transfer. The computer control system then activates the relevant control loop, including its pumps and valves, to execute this command. Manual adjustments and overrides to automatic functions are processed through the digital control system. All process functions are controlled through the automated control system.

The operator uses the console to input process parameter set points, to respond to prompts for requested data and process alarm conditions, and to order hard copies of CRT displays and logs. Operators interact with the system through numerical keys, dedicated keys, and "soft" keys whose functions are programed to access specific CRT displays. In addition, each console has a four-position key-lock switch. The four positions and their functions are as follows:

| Position | Function |
| --- | --- |
| • LOCKED | Allows all displays to be monitored only. |
| • OPERATE | Allows all normal operating functions to be performed. |
| • TUNE | Allows certain process parameters or variable control loop parameters to be changed. |
| • CONFIGURE | Allows graphic displays, control sequences, control logic, etc., to be modified or generated. |

Although the description of the process control system in the facility license stated that keys for the lock-switch are controlled by the shift supervisors, process engineer, and control system engineer, control room operators now maintain control of the key since the shift supervisor position was eliminated in March of 1990.

The CONFIGURE mode also allows access to numerous diagnostic and troubleshooting programs. Operations in the CONFIGURE mode can only be made at a separate engineering keyboard in the control room or at a console located in the computer room by designated personnel, such as the control systems engineer, a designated assistant control systems engineer, or the process engineer.

Process operations start with a series of prompts displayed on the CRT for parameter data input to begin batch processing. Following input of initial data and once a process has begun, the operator monitors the process and provides the requested data inputs as prompted by the automated system. For solvent-extraction (SX) operations, operators typically monitor several displays of different portions of the system (e.g., extraction, scrub and strip column parameters) on one CRT and trends of selected process variables on another. The other console is used to monitor the other URU processes (radwaste, nitrate waste and fluoride waste) by another operator.

The SX system does not contain an automatic shutdown function. SX shutdowns must be initiated by an operator. However, the failure of certain SX system components will cause effected parts of the system to sequence automatically to a problem phase. For example, a problem phase in the extraction column portion of SX will result in the shut down of specific pumps and valves to isolate the inlets and outlets to the three extraction columns.

### 6.1.2 System and Personnel Performance During the Incident

Prior to the evening of May 28, 1991, the SX process had been running steadily for about six weeks. At about 8:30 p.m. on May 28, 1991, the trend display of density for Column A, denoted as 1-DIC-300 on Figure 6.2, and the Column A interface level value displayed on the SX A, B, and C column screen (Figure 6.3), both indicated values below their respective normal range values (i.e., 0.85 -- 0.92 for density and approximately 65 percent for interface level). The density trend updates, recorded at four-minute intervals, indicate that the Column A density values dropped below 0.80 at approximately 8:30 p.m. The high and low alarm setpoints for the Column A density and interface parameters were set to the maximum upper and minimum lower range values so that they were essentially disabled. At approximately 9:30 p.m., control room operator A (CRO A) completed feed make-up operations, returned the console display to the trend of Column A density, and noticed that the value was below 0.80. He also noticed that the aqueous feed (AF) rate had increased from 180 liters/hour to 220 liters/hour. CRO A reduced the AF rate back to 180 liters/hour.

Parameters with alarm capabilities are visually displayed on CRT screens but are not audible. Further, there is no distinction between a process parameter alarm and an alarm that is important to nuclear criticality safety. Originally, the operator interface system was designed to provide both visual and audible alarms. However, the audible alarm capability was removed entirely because operators complained that some alarm set points were too close to normal operating ranges and were a nuisance when they actuated frequently. A parameter alarm is displayed by a change in its normal state (from steady to flashing) and color (from the default color to red). Parameter alarms are displayed on the bottom of all CRT screens, in addition to being shown on the screen where the status of that parameter is normally displayed. Thus, operators viewing any console screen would be aware of any indications of parameter alarms. However, because all the audible alarms are inoperative, operators must actually be viewing console screens before they are aware of alarm conditions.

After CRO A reset the AF rate, he called an operator assigned to the SX operations floor to check on Column A parameters. A short time later, the floor operator called CRO A to tell him that level control valve LCV-300 did not appear to be responding to control demands. CRO A attempted to close LCV-300 from the control room console but the valve did not respond. CRO A then instructed the floor operator to close a manual block valve upstream of LCV-300. The manual block valve is used to isolate Column A from the aqueous waste line during shut down conditions. This manual valve, like most others in the URU, does not have a unique identifier or label. Much of the equipment in the facility is not labeled. Operators stated that they learned the location and functions of components and equipment through on-the-job training. With the manual valve closed, CRO A stated that Column A interface level and density values began to rise into their normal operating ranges. Evidence of this change in density can be seen on Figure 6.2, where at approximately 11:00 p.m., the density value rose to approximately 0.92, remained relatively steady for about 12 minutes, and then dropped below 0.80. CRO A did not notify maintenance personnel of LCV-300 problems, nor was the area coordinator on shift notified that there were SX problems until 10:30 p.m. The problem was reported to the relief shift's area coordinator during shift turnover at 11:00 p.m.

SX operations continued in this configuration until shift turnover at 11:00 p.m. The relief or oncoming CRO A noticed Column A interface and density problems and questioned the outgoing CRO A about SX system status. After hearing of the problems and attempts to correct them, the relief CRO A instructed the floor operator to open the manual block valve upstream of LCV-300. Before leaving his shift, outgoing CRO A took a sample of the contents in an aqueous waste quarantine tank from the URU lab sample sink (Figure 6.4) and noted to the relief CRO A that it was clear, which he felt was an indication of no significant uranium concentrations, and concluded that there was not a problem with the SX system. The oncoming CRO, now in charge of the shift, notified maintenance workers to troubleshoot and repair LCV-300 (there is no maintenance supervision during non-day time shifts and week ends). CRO A also called the process engineer at home to tell him of the problem with the SX system. After some discussion, they agreed that placing the SX system in a "warm start-up" sequence might cause LCV-300 to reposition and function normally again. Warm start-up is used to restart the SX system from a temporary shut down sequence during which Column A process parameters are re-established. However, the attempted warm start-up did not cause LCV-300 to re-position. CRO A also tried unsuccessfully to get LCV-300 re-positioned by resetting its open and close control set points in the process control computer.

For the next six hours, SX operations continued with LCV-300 not functioning and with the upstream block valve being opened and closed in an attempt to keep the SX system operating. While maintenance personnel attempted to repair LCV-300, they removed air pressure from the valve actuator/positioner and the valve failed closed as designed. With LCV-300 closed, the solvent and organic feeds continued to fill Column A to the point of flooding. Flooding Column A causes the solvent/organic mixture to overflow into Column B. At this point, the SX process control system automatically isolates the columns (problem phase) by securing all inlet and outlet valves and pumps. Knowing that the contents of the flooded Column A have to be transferred to the V-290/V-291 aqueous waste (AW) quarantine tanks to bring the SX process back to normal operating conditions, CRO A transferred the contents of V-290 and V-291 seven times to waste accumulation tank V-103 during a six-hour period using the TUNE mode (see Table 6.1). Transfer operations continued until a lab analysis of a quarantine tank sample indicated 6977 ppm uranium, at which time CRO A called the process engineer at home again to explain the situation. The process engineer told the CRO A to shut SX operations down. When the team asked if it would have been a problem to shut the SX system down and allow the contents of columns A and B to remain as is, CRO A stated that it would not have been a problem and in retrospect it would have been the right thing to do. However, CRO A indicated that in order to keep the SX process running and avoid shutting it down, he had hoped that maintenance personnel would have repaired LCV-300 quickly enough to re-establish normal SX operations. He also indicated that starting the SX process from a shut down condition was a rather slow process and could take as long as 24 hours to reach a steady state operation without having any problems.

The contents of aqueous waste quarantine tanks V-290 and V-291 (600-gallon capacity, each) may be transferred to either waste treatment accumulation tank V-103 (20,000-gallon capacity) or to rework tank V-225 (600-gallon capacity). By procedure, AW quarantine tank contents may be transferred to V-103 if the uranium concentration is 150 ppm or less; otherwise, it must be transferred to V-225 for reprocessing. By procedure, the uranium concentration in a quarantine tank is determined in the following manner. With the SX control system in the operate or automatic mode, quarantine tank contents begin to recirculate when the level reaches 50 percent. The quarantine tank recirculation system includes a lab sample loop. At the 90-percent full level, the inlet valve to the tank closes automatically and the contents continue to recirculate for an additional 15 minutes. This 15-minute recirculation is to assure that the sample taken is an adequate representation of the tank's contents. After 15 minutes of recirculation, the operator receives a prompt on the CRT screen for the lab sample analysis results from a lab sample taken from the filled tank. At such time, the operator requests that the lab analyst take a sample and analyze it for uranium concentration. Once the lab results are obtained, the operator inputs the results into the process control computer and, depending on the values, the quarantine tank's contents are discharged to either V-103 or V-225.

The original design of the automated sample system included provisions for it to automatically take a sample, analyze it, and then send the results to the process control system computer. The process control system computer would then compare the lab sample analysis results with prescribed values and transfer the contents of the sampled quarantine tank to the appropriate accumulation tank for further processing. During the incident, however, the automated lab analysis system was not operable. Under these circumstances, the control system computer defaulted to the operator prompt mode previously described. The operator may, however, circumvent the automatic controls by placing the console in the TUNE mode and request the computerized process control system to transfer the quarantine contents to either V-103 or V-225. In the TUNE mode, the operator may transfer the quarantine tank contents before the tank level reaches the 90-percent mark without recirculating the contents for 15 minutes and without having to input the lab sample analysis results. During interviews with other URU operators, they stated that if the quarantine tanks had a concentration greater than 150 ppm of uranium and V-225 was not available (e.g., was already full), that the quarantine tank's contents could be dumped to the floor to drain into the nitrate waste treatment system sump. The operators further stated that if the acidic fumes from the quarantine tank contents being dumped on the floor were too strong to withstand, water would be sprayed on the drainage to reduce the fumes. Dumping quarantine tank contents to the floor is not governed by any written procedures nor was management aware of this practice. However, a review of the operator's logs for May 26, 1991, indicated that the contents of V-291 (540 gallons at 187 ppm of uranium) was dumped to the floor because V-225 was about 50-percent full, thus confirming the practice.

During the 11:00 p.m. to 7:00 a.m. shift on May 28-29, 1991, eight transfers from V-290 and V-291 were all performed in the TUNE mode. During interviews with the lab analyst and CRO A on that shift, the team learned that CRO A made several trips to the lab sample sink (Figure 6.4) to visually examine the quarantine tank contents. The URU lab is adjacent to the control room (Figure 6.5). CRO A stated that he was visually analyzing quarantine tank samples to determine if there were high uranium concentrations. CRO A stated that the visual analyses appeared to indicate low or expected uranium concentrations and he subsequently transferred the V-290 and V-291 contents to V-103. The lab analyst stated that samples were taken and analyzed only when requested by CRO A and that this information was recorded on the lab logs. A review of these lab logs indicated that, of those samples analyzed, three were found to have uranium concentrations of less than 150 ppm and one had a value of 6977 ppm, at which time CRO A shut the SX system down and transferred the contents to tank V-225.

During interviews with URU operators, the team learned that sample analyses of the contents of V-290 or V-291 were often performed outside the actions prescribed by procedure. Operators would request sample analyses as the tank was still being filled in order to expedite operations. By having the lab sample analysis performed while the tank was filling (and possibly completed by the time a tank reached the 90-percent level), the contents could be transferred shortly thereafter to keep the process running continuously. Operating this way negated the need to recirculate a tank 90-percent full for 15 minutes prior to taking a sample for analysis. Operations in this manner were performed with the process control computer in the TUNE mode. Some operators estimated that the TUNE mode was used half of the time during process operations. Because of insufficient oversight of operations, plant managers, engineers, and planners were not aware of the extent of this practice.

The team's review of the operator's log sheets and the lab analyst's log sheets indicates that the correlation between sample analysis result times and operator's recorded times of quarantine tank transfers were inconsistent, and that information as to where the quarantine tank's contents were transferred was not found. The team learned during interviews with the plant manager and production management engineers that they believe there were no requirements for keeping control room or lab logs, no criteria for logging lab sample times, no criteria for logging where quarantine tank contents were transferred, no requirements for maintaining computer data logs, and no procedures or instructions on the maintenance of logs, or of how and what information should be logged. However, the procedure governing URU process lab administrative practices clearly states that the results of all measurements should be reported to the CRO and logged as soon as the results are available. Because there is no requirement for operators to maintain logs and because of the inconsistent manner in which records were maintained and controlled, auditing compliance with AW transfer criteria was not possible even if management were inclined to do so, which they were not. In fact, according to the Acting Manager, NFCM, there is no Quality Assurance oversight of the URU process.

The operator's anxiety to keep the process running continuously may be attributed in part to faster filling rates for the quarantine tank after the addition of several condensate system discharge lines to the aqueous waste quarantine tanks inlets. These lines were installed and placed into service around May 1987 and November 1990. Normal AW flows are about 50 gallons per hour; the condensate lines add approximately 345 more gallons per hour. (185 gallons per hour from the May 1987 change and 160 gallons per hour from the November 1990 change.) The technical analyses in the Facility Change Request (FCR) for the system performed prior to the change did not include an evaluation of the impact of adding the condensate flows on SX operations and the challenge it presents to the operator. In addition, nuclear criticality safety audits focus on equipment changes and not on changes to plant operations.

It is probable that the faster filling times of the quarantine tanks were also a reason why operators have emptied the tank's contents into the nitrate waste sump, as previously mentioned. In situations where the appropriate aqueous waste recipient tank did not have adequate volume to allow transfer from a quarantine tank, and the operator wanted to keep the SX system operating continuously, then dumping to the nitrate waste sump was a means of accomplishing that goal.

## 6.2 Operating Procedures

The primary operating procedure used to run the solvent extraction and aqueous waste quarantine system provides operators with instructions and guidance and with lists of related procedures, equipment, special instruments and control systems and valves. The procedure contains the instructions for start-up, normal operations, shut down, abnormal conditions, basic maintenance, industrial safety, and troubleshooting. The procedure also contains examples of operator interface system displays, relevant nuclear safety release/requirements (NSR/R), operating parameter sheets, environmental protection requirements, and functional test instructions. The procedure includes, in sequence, both automated actions and required operator actions. Required operator actions are distinguished from automated actions by bold print.

In general, the procedure did not emphasize a "safety first" approach to operations but rather promoted a continuous operation philosophy. It is uncertain for several reasons whether compliance with the SX system operating procedures would have prevented or even mitigated the consequences of the incident under investigation because of ambiguity in the abnormal conditions instructions:

(1) The abnormal conditions section of the procedure instructs the operator to refer to the troubleshooting section for failure of an interface control loop, such as LIC-300 (Column A), which contains the controls for LCV-300. The troubleshooting section instructs the operator to check for closed valves, blocked piping, failed instruments, leaky level control valves (e.g., LCV-300 for Column A), or pump operations, and contains a reminder that some loss of level control is normal during start up and, under those conditions, engineering may be contacted. There is no mention of shutting the SX system down, yet another part of the procedure states that the SX temporary shutdown mode is to be used for troubleshooting activities. However, the detailed procedures do not specify that the system be shutdown before troubleshooting begins nor the point at which troubleshooting would mandate a system shutdown.

(2) Instructions for other abnormal conditions direct the operator to contact maintenance, instrumentation, or engineering, except in one case: if organic extractant is found in an aqueous waste sample from the recirculation loop, the operator is to shut the SX system down. No other cases of abnormal operations require SX operations to be terminated, even though a senior process engineer stated that the SX system should not be operated without a functioning level control valve.

(3) Instructions for aqueous waste quarantine operations describe the automated process of filling, recirculating, sampling, and transferring the contents of aqueous waste tanks V-290 and V-291. Required operator actions include allowing the contents of the tanks to recirculate for 15 minutes before asking for a sample analysis. However, the decision as to which tank receives the contents of an aqueous waste tank is made by the operator. There is no mention of having an independent source verify the transfer. Nor are there any contingency actions for situations in which a tank is not available for aqueous waste transfer. Also, the procedure does not contain any instructions or guidance for dumping the contents of an aqueous waste tank into the nitrate waste system sump.

## 6.3 URU Operator Training

The licensee's initial training for URU operators during the start-up of URU operations from 1984 to 1986 was comprehensive. For approximately two and one half years, operators, supervisors, managers, engineers, lab operators, and maintenance personnel received formal classroom, vendor, and on-the-job training for many aspects of URU operations. Operators received training on the URU process control computer, process theory and operations, nuclear safety release/requirements (NSR/R), and procedures. In addition, operators trained on a process control console simulator for normal and abnormal conditions and were involved with the verification of operating procedures. The retraining plan specified that the shift supervisor was to perform on-the-job training on new or revised procedures on an annual or as-issued basis. Since the shift supervisor position was eliminated in 1990, the area coordinators and process engineers have assumed responsibility for training operators on the use of new or revised procedures.

Most of the A and several of the B CROs that received the initial training were still employed at GE NFCM as such at the time of the incident. The CRO on shift from 3:00 p.m. to 11:00 p.m. on May 28 became a URU operator during the late stages of the initial training period. The CRO on shift from 11:00 p.m. through 7:00 a.m. on May 29, 1991, was a URU operator during initial operator training.

After initial training was completed, there was no formal training program established for new URU operators and retraining for existing operators other than general nuclear safety orientation and measurement training. However, the team learned that a new training program was being developed for plant personnel. The emphasis will be on training that is directly related to the needs of the operators as it relates to their job requirements. Currently, new operators only receive on-the-job training from experienced A operators. The area coordinators are responsible for the management of on-the-job training. URU area coordinators are required to have enough knowledge of the processes and products to be able to address specific problems. The area coordinators on staff at the time of the incident were not working at the URU during the initial training period and have not received any formal URU process training. During interviews with several area coordinators, the team learned that they had been in those positions for approximately one year and that they rely on the operators to inform them of process operations and problems they may encounter. They all purport to have some knowledge of URU operations but not to the extent where it would be helpful in assisting operators during abnormal conditions.

## 6.4 Supervisory and Management Oversight

The management structure at the GE NFCM is organized in such a manner that there is no single organization that has sole responsibility for URU. Managers and support organization personnel have broad areas of responsibility and large spans of control. As illustrated on the organizational charts in Figures 6.6, 6.7, and 6.10, managers with overall responsibility for the NFCM, the Fuel Manufacturing Operation (FMO), and the chemical process areas in the FMO are responsible for many oversight functions in addition to the URU. The simplified organization chart in Figure 6.11 illustrates how the Manager, NFCM, is the only person who has oversight responsibility for all URU operations and support organizations. Figures 6.8 and 6.9 illustrate that support organizations at GE NFCM are not structured to provide dedicated services to any one particular operation, such as the URU. This management and supporting organizational structure created a lack of ownership for any one area of the facility.

URU operators were concerned that because managers and area coordinators were often so busy with their many areas of responsibility that they could not effectively oversee operations. The operators stated that they believe that is the most likely reason why managers did not make regular walk throughs of the process and production areas. Likewise, the other support personnel had to attend to the demands of the other areas for which they provided oversight and support functions.

In March 1990, the management structure at GE NFCM was changed. Supervisory positions for those who had direct responsibility for oversight of the URU were eliminated. The area coordinator positions were established at that time to encompass the responsibilities previously held by the supervisors. The chemical area coordinators have responsibility for the URU and the $UF_6$ conversion area on the day shifts during the week and the fuel support, pellet production furnace, and the gadolinia furnace on non-day shifts and all shifts during weekends (Figure 6.10). As part of the March 1990 change, CRO A operators were given the responsibility of overseeing the day-to-day operations of the URU.

In addition to the lack of ownership described above, there did not appear to be a "safety first" attitude communicated throughout the facility. URU operators meet with their area coordinators once a week at an "11-minute meeting" to discuss the general status of the facility and a weekly uranium recovery production report. Operators noted that meetings with upper management emphasized production goals without appropriate qualifications for safety considerations. Management stressed that production rates needed to be higher because the fuel fabrication industry was very competitive and that the survival of the company was dependent upon the success of meeting production goals. During the entrance and exit meetings with the licensee, the team noted that GE managers focused their attention on recovery from the incident and on restart activities so that they could begin to meet their customers' needs.

Through interviews, the team has determined that licensee management expects area coordinators to focus their attention where it's needed most, to verify the adequacy of operator and system performance, to coordinate and prioritize maintenance activities as necessary, to have the requisite knowledge and experience of chemical processes, and to have good communications and people skills. Operators, likewise are expected to run the processes according to procedures, point out problems with procedures, train and advise less senior or new operators, perform some maintenance and housekeeping, maintain records, and to be capable of resolving problems and not just reporting them. However, these expectations were not being fulfilled.

The team determined that several weaknesses in the supervisory and management oversight of the URU contributed to the potential seriousness of the incident:

- Area Coordinators serve as the only direct management during non-daytime shifts and on weekends. They have a broad area of responsibility and a large span of control.

- There was no technical support and maintenance supervision on site during non-daytime shifts and on weekends.

- There is no distinction between process parameter and criticality safety parameter alarms in the control room.

- A "safety first" attitude was not communicated throughout the facility.

- Management's presence in the process and production areas was limited.

- There is no quality assurance oversight of URU activities.

- Nuclear criticality safety audits to assure that criticality controls were implemented as intended focus on equipment changes and not operations.

- Management was not aware of the extent that the computer control system was being operated in the TUNE (manual override) mode.

- Management was not aware of deviations from aqueous waste sampling and transfer procedures. There were no audits of compliance with procedures.

- Operator performance is not measured or observed within a training/qualification program after initial training during URU start-up in 1984 through 1986. New operators receive only on-the-job operations training from senior operators and area coordinators. Nor was there requalification training for operators.

- For abnormal conditions, operating procedures focused on continued operations rather than on bringing the process to a safe condition.

- Changes and modifications have been made to plant equipment and systems without an adequate evaluation of their effect on operations.

- Management meetings with plant personnel emphasized production goals without appropriate qualifications for safety considerations.

## 6.5 Findings and Conclusions

This section summarizes the predominant findings and conclusions from the team's evaluation of human factors issues related to the incident.

The management organizational structure at GE NFCM dictated a wide span of control such that no single group had sole responsibility for the URU. The many peer organizations such as operations, engineering, maintenance, chemistry, instrumentation and calibration, and regulatory compliance all report to the Manager, NFCM. The Manager, NFCM, had 12 direct and 4 indirect organizations reporting to him. This situation resulted in limited management presence in the process and production areas, a lack of ownership of the URU and assurance that activities were conducted in a safe manner.

Area coordinators' broad area of responsibility made it difficult for them to provide adequate supervisory oversight especially when they serve as the only direct management during non-daytime shifts and weekends. Without direct management oversight and guidance, operators had the flexibility to do whatever necessary to meet management's expectations.

Management meetings with plant personnel emphasized that production was the facility's overriding concern. Similarly, operating procedures emphasized continuous operations rather than stressing safe practices. Modifications to the computerized process control system served to promote production and eliminated features which were intended to support safe operations such as alarms, interlocks and sub-system shutdowns. These changes were made without an adequate evaluation of the impact on criticality safety. Those alarms that were not eliminated did not convey any distinction between process parameters and criticality safety parameters.

Operator's prevalent attitude was to maintain continuous operations to the extent that procedures and automatic controls for the criticality control sampling system for SX aqueous waste (AW) were circumvented. The operators anxiety to keep the process running may have been attributed in part to modifications in another part of the SX system that resulted in significantly increased inlet flow rates to AW tanks. These modifications did not include an evaluation of the impact the increased flow rates had on operators and process operations.

Management's limited cognizance of URU activities were manifested in several ways. Routine evaluations of operator performance relative to procedural adherence and practices were non-existent and operators were questioned only when productivity was affected. The absence of requirements for control room operators to maintain logs made audits of operations nearly impossible, even if management chose to do so. Nuclear criticality safety audits performed to assure that criticality controls were implemented as intended focused on equipment changes and not operations.

Operators exhibited ingenuity regarding production. For example, operators would on occasion, dump AW quarantine tank contents with high uranium concentrations into a sump. This practice may have been contributed in part to deficient operating procedures. The governing AW transfer procedure did not contain any contingency actions for situations when uranium concentrations in the AW quarantine tanks were higher than limits for transfer to the unfavorable geometry waste accumulation tank and the aqueous waste rework tank volume was not sufficient or available.

Table 6.1 Quarantine tanks V-290 and V-291 sample analyses and contents transfers
May 28-29, 1991

| Tank No. | Tank Level At Analysis Completion | ppm U and Time of Sample Analysis | Time Contents Transferred | Tank Level at Transfer | Tank Contents Transferred To: | Manual or Automatic Transfer |
|---|---|---|---|---|---|---|
| V-290 | 90 % | 74 @ 8:50 p.m. | 9:35 p.m. | 90.5% | V-103 | Manual |
| V-291 | 78% | 69 @ 10:00 p.m. | 10:15 p.m. | 90% | V-103 | Manual |
| V-290 | — | Unmeasured | 11:00 p.m. | 90% | V-103 | Manual |
| V-291 | 50% | 40.1 @ 11:30 p.m. | 12:10 a.m. | 90% | V-103 | Manual |
| V-290 | 80% | 42 @ 1:00 a.m. | 1:05 a.m. | 89.5% | V-103 | Manual |
| V-291 | — | Unmeasured | 1:30 a.m. | 39% | V-103 | Manual |
| V-290 | 5% | 48.7 @ 1:30 a.m. | 2:55 a.m. | 90% | V-103 | Manual |
| V-291 | — | Unmeasured | 4:00 a.m. | 81% | V-103 | Manual |
| V-290 | — | Unmeasured | 4:30 a.m. | 64% | V-225 & V-103 | Manual |
| V-291 | 58% | 6977 @ 5:10 a.m. | 5:20 a.m. | 80.5% | V-225 | Manual |

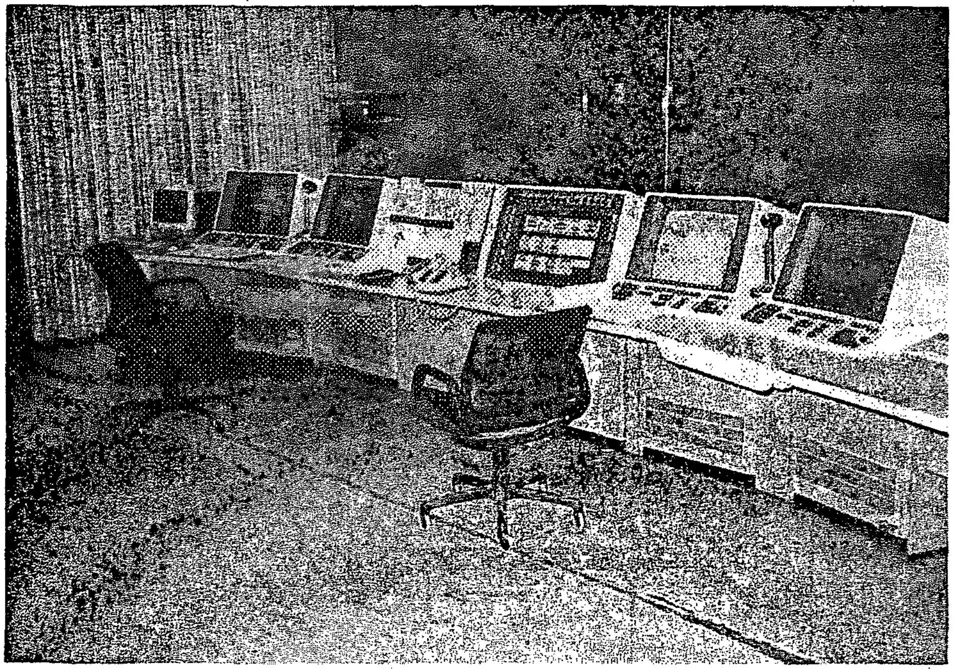

Figure 6.1 Uranium Recovery Unit (URU) control room

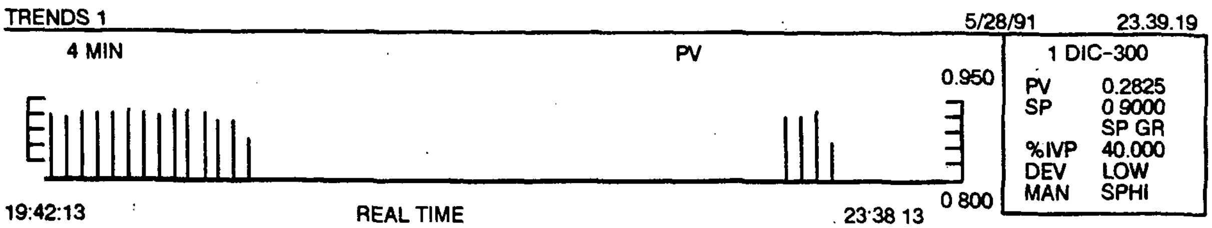


Figure 6.2 URU trend display of column A density

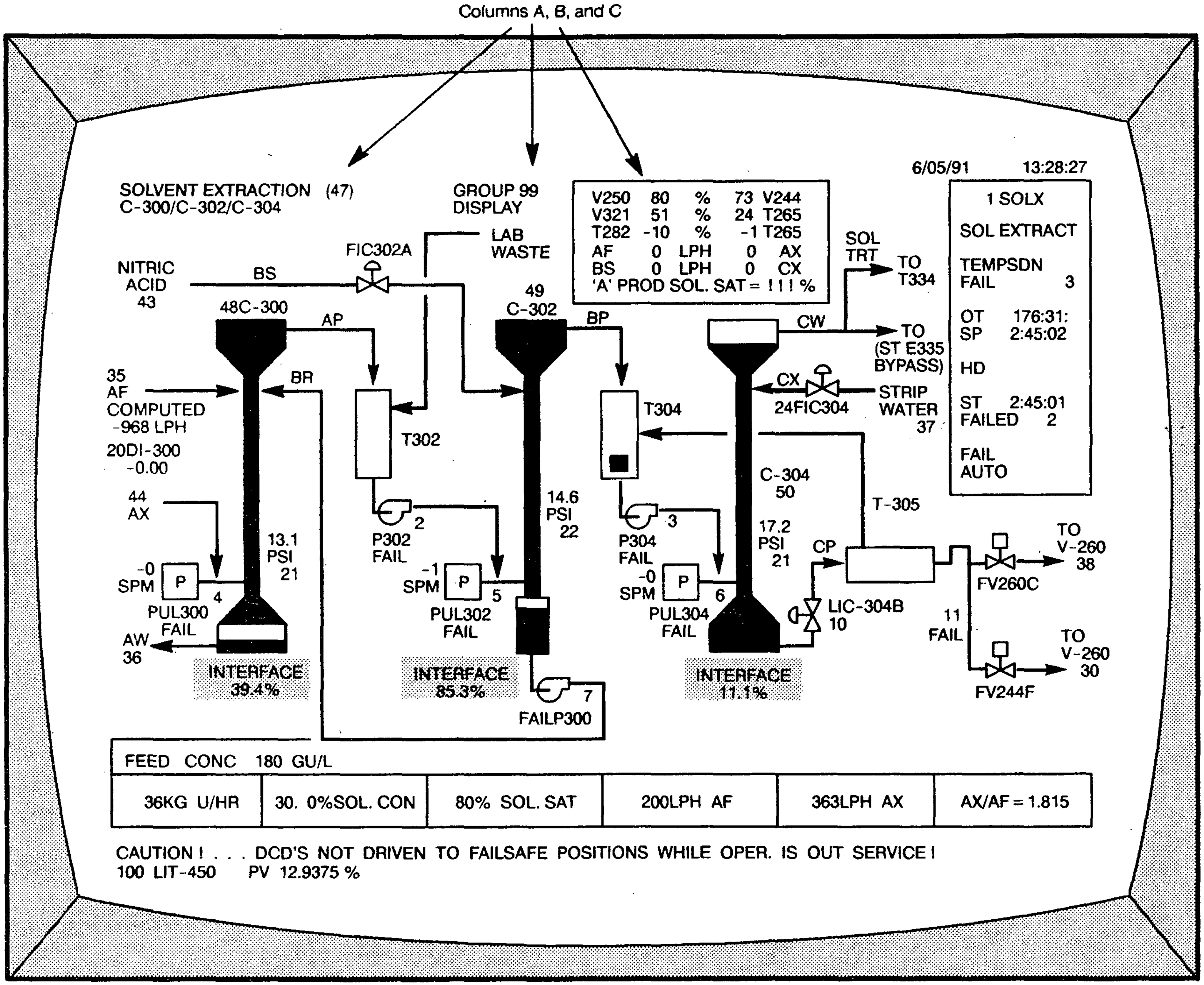


Figure 6.3 Control computer display for columns A, B, and C

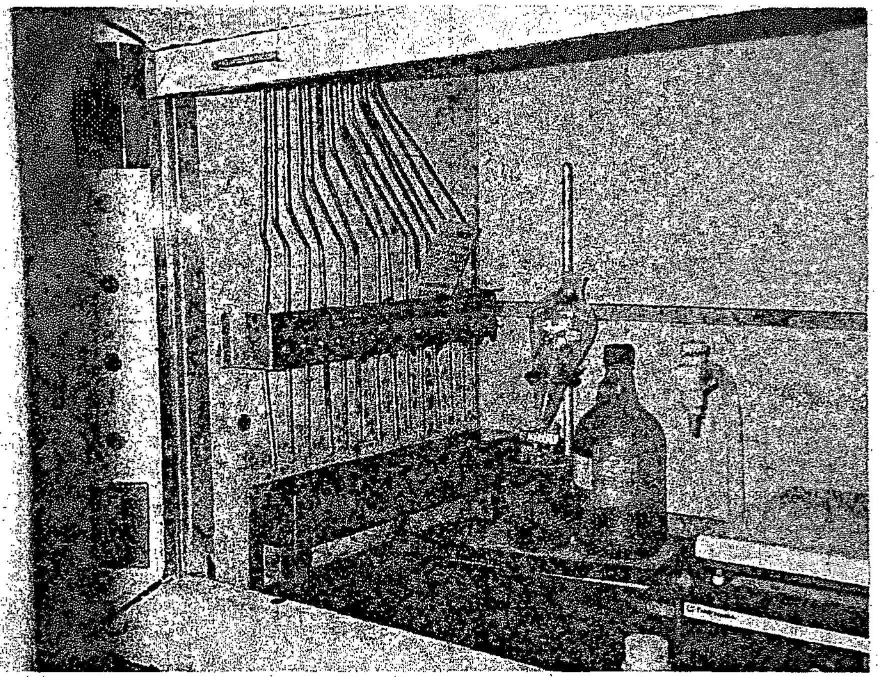

Figure 6.4 URU laboratory sample sink

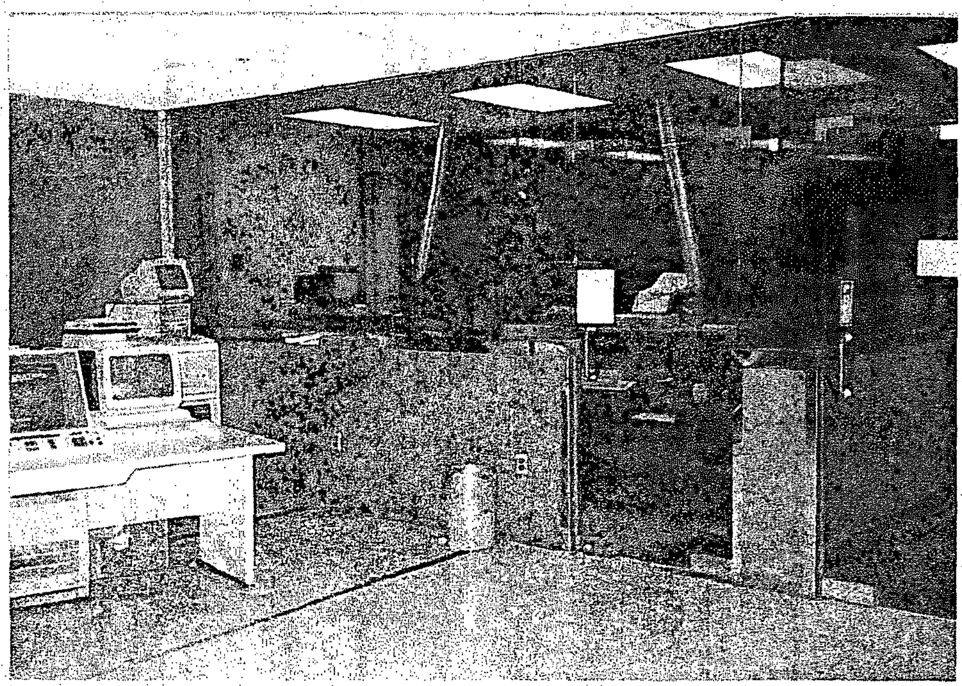

Figure 6.5 URU laboratory's location adjacent to the control room

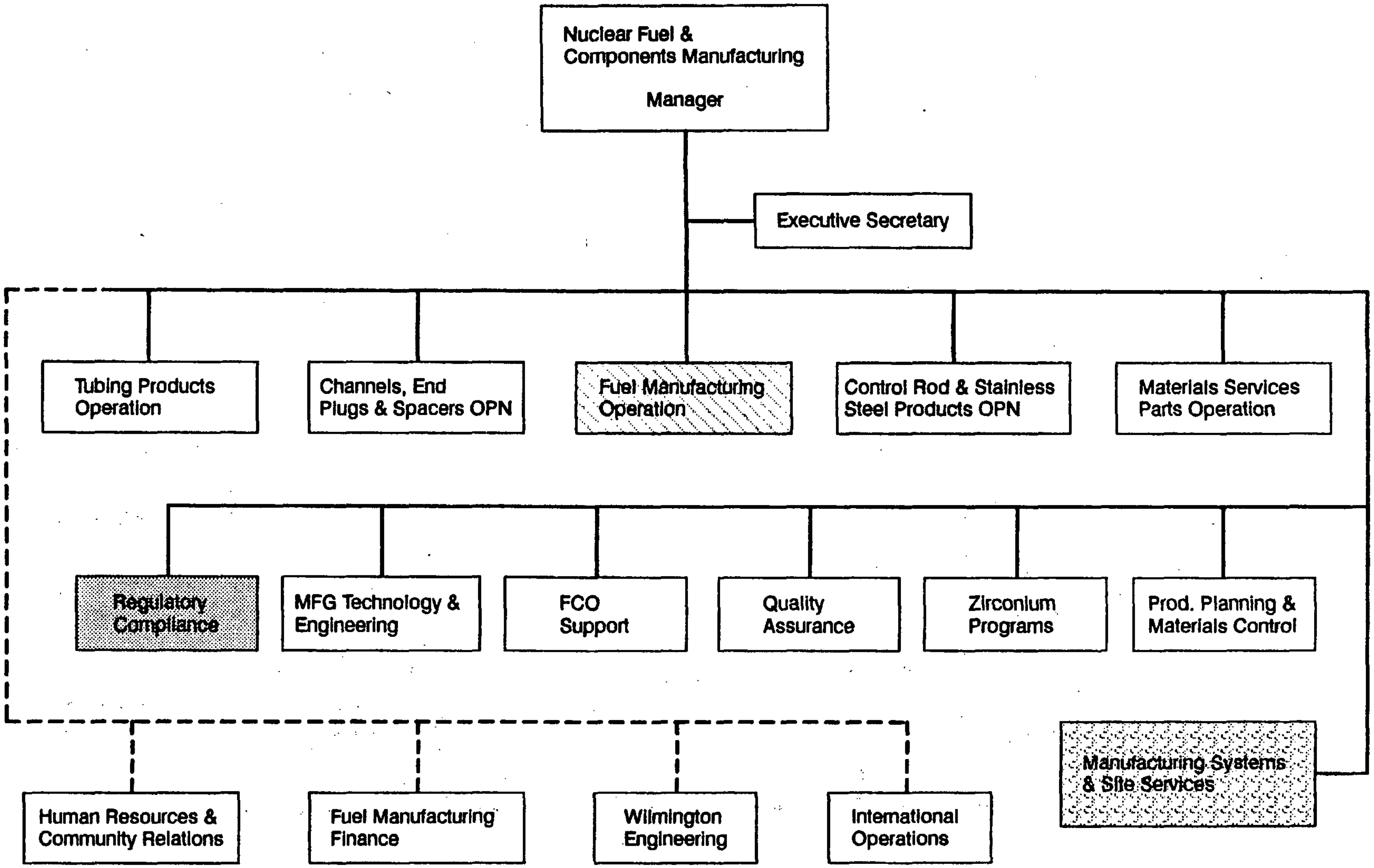


Figure 6.6 Nuclear Fuel & Components Manufacturing Facility organization chart

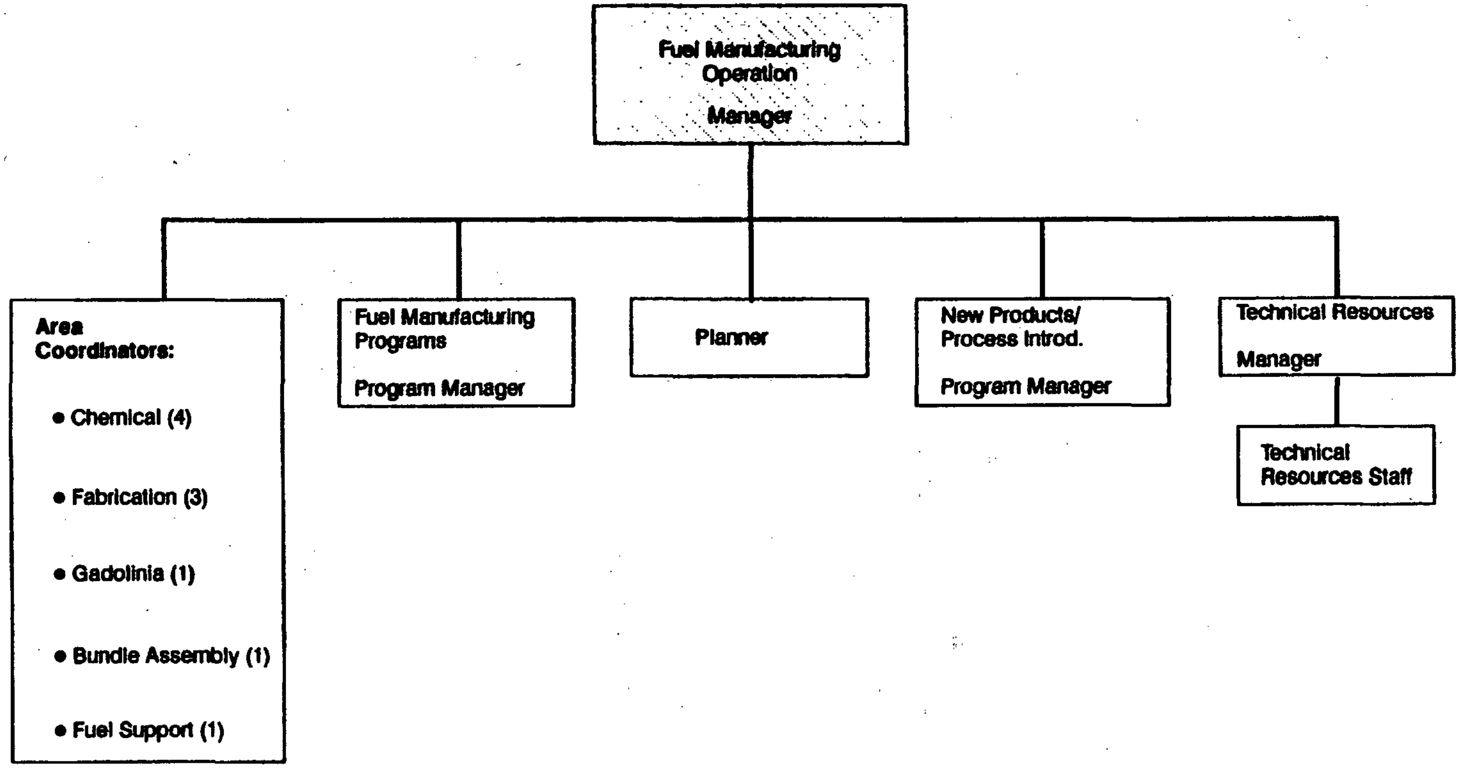


Figure 6.7 Fuel Manufacturing Operation organization chart

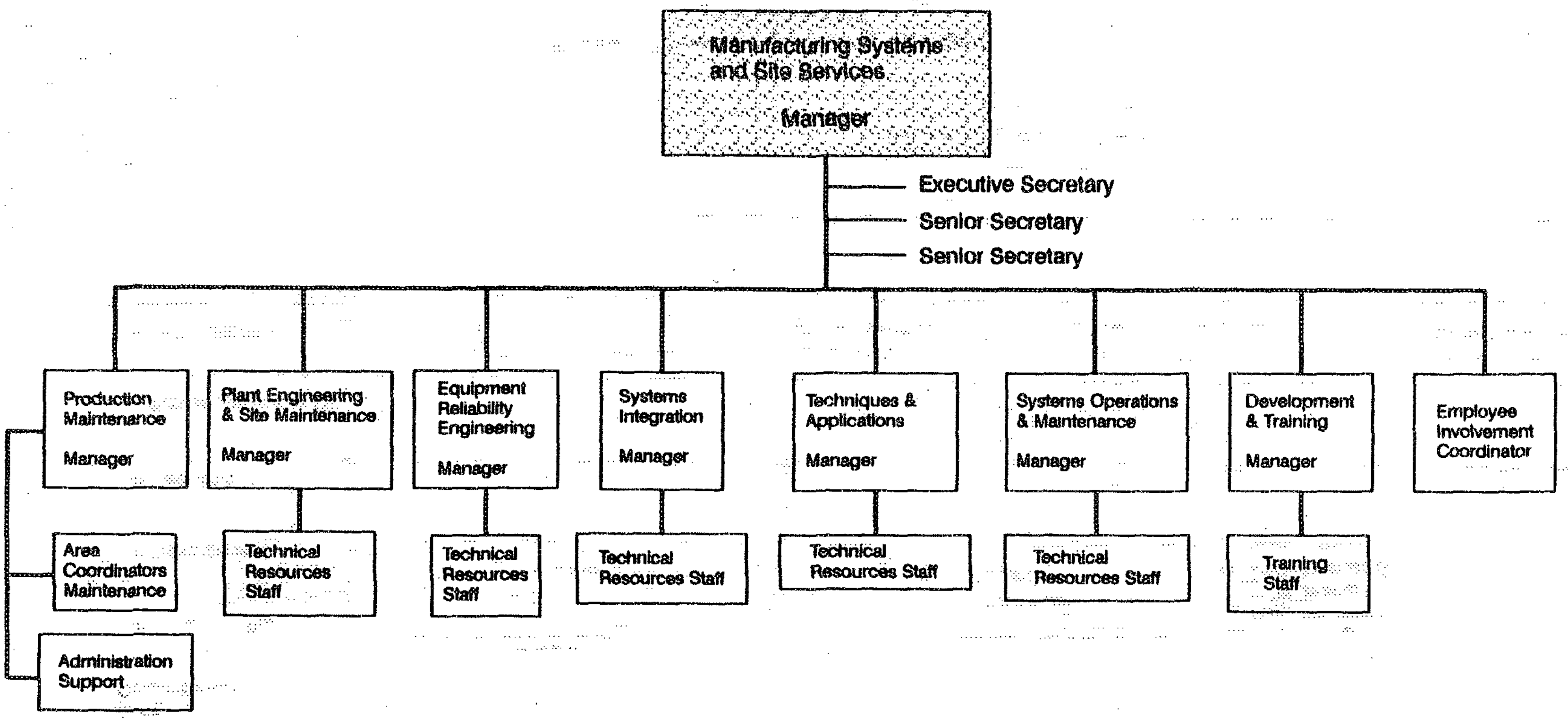


Figure 6.8 Manufacturing Systems and Support Services organization chart

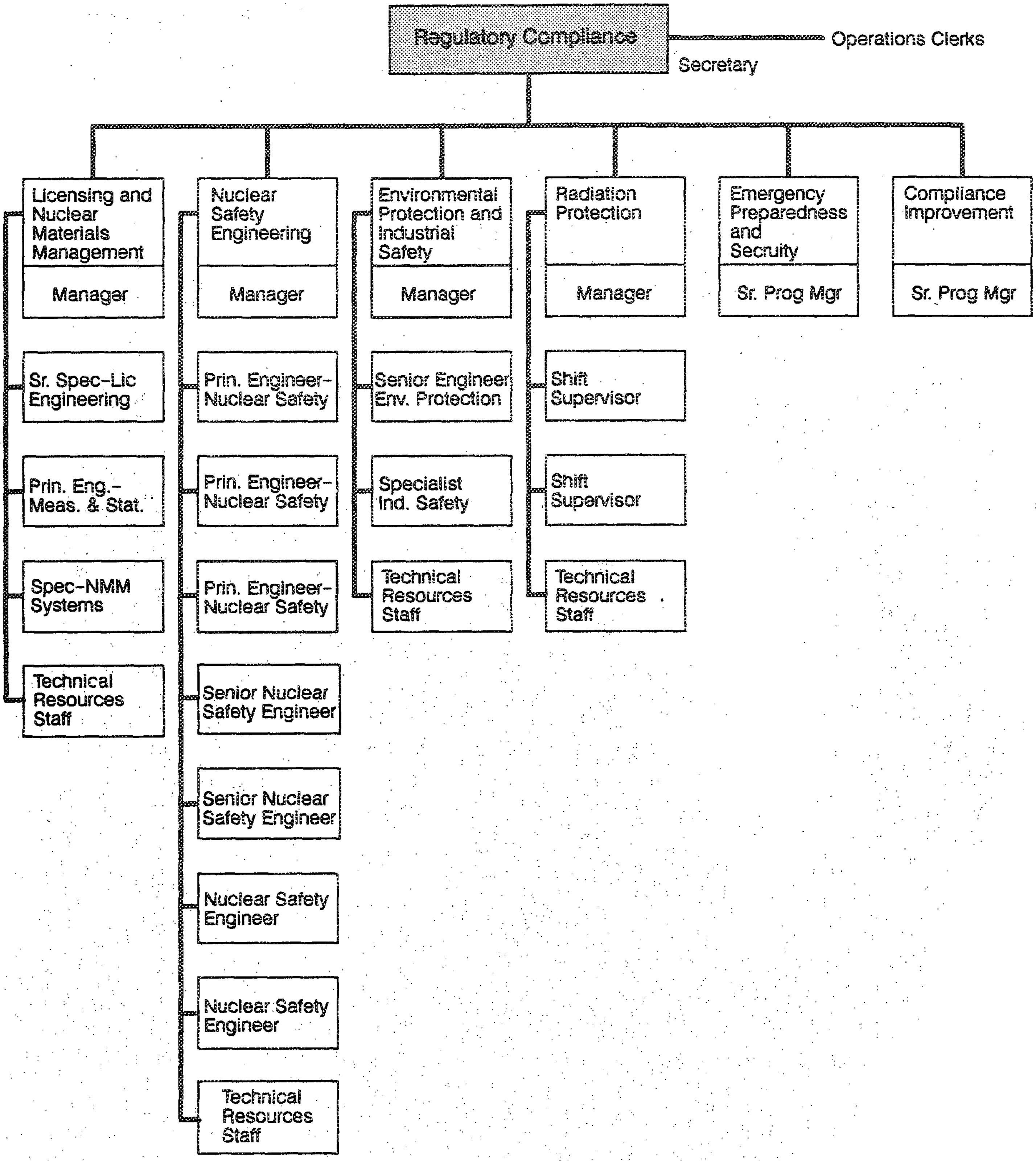


Figure 6.9 Regulatory Compliance organization chart

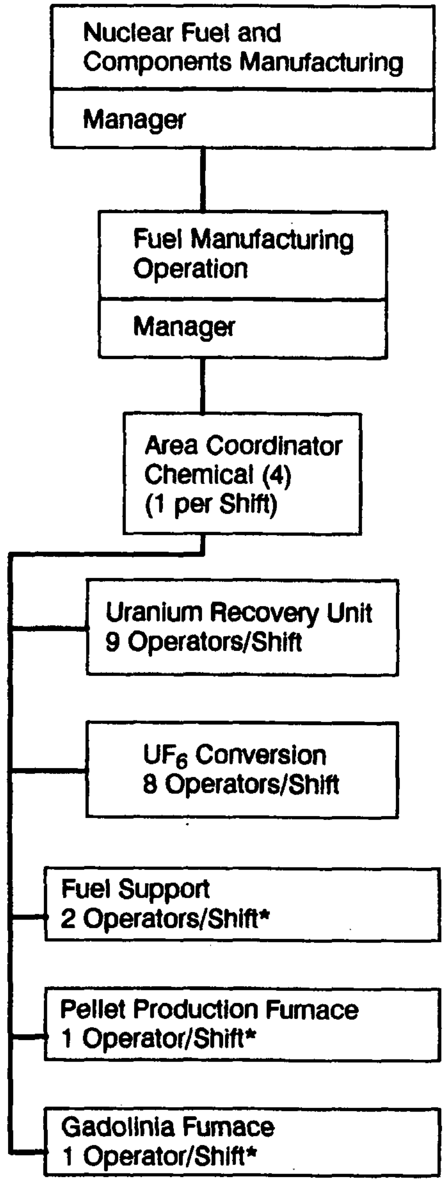


Figure 6.10 Managerial oversight for chemical operators on each shift (21 operators covering an area of approximately 21,000 square yards and encompassing several buildings and tankage areas)

* These operators are supervised by one chemical area coordinator during non-day shifts and weekends.

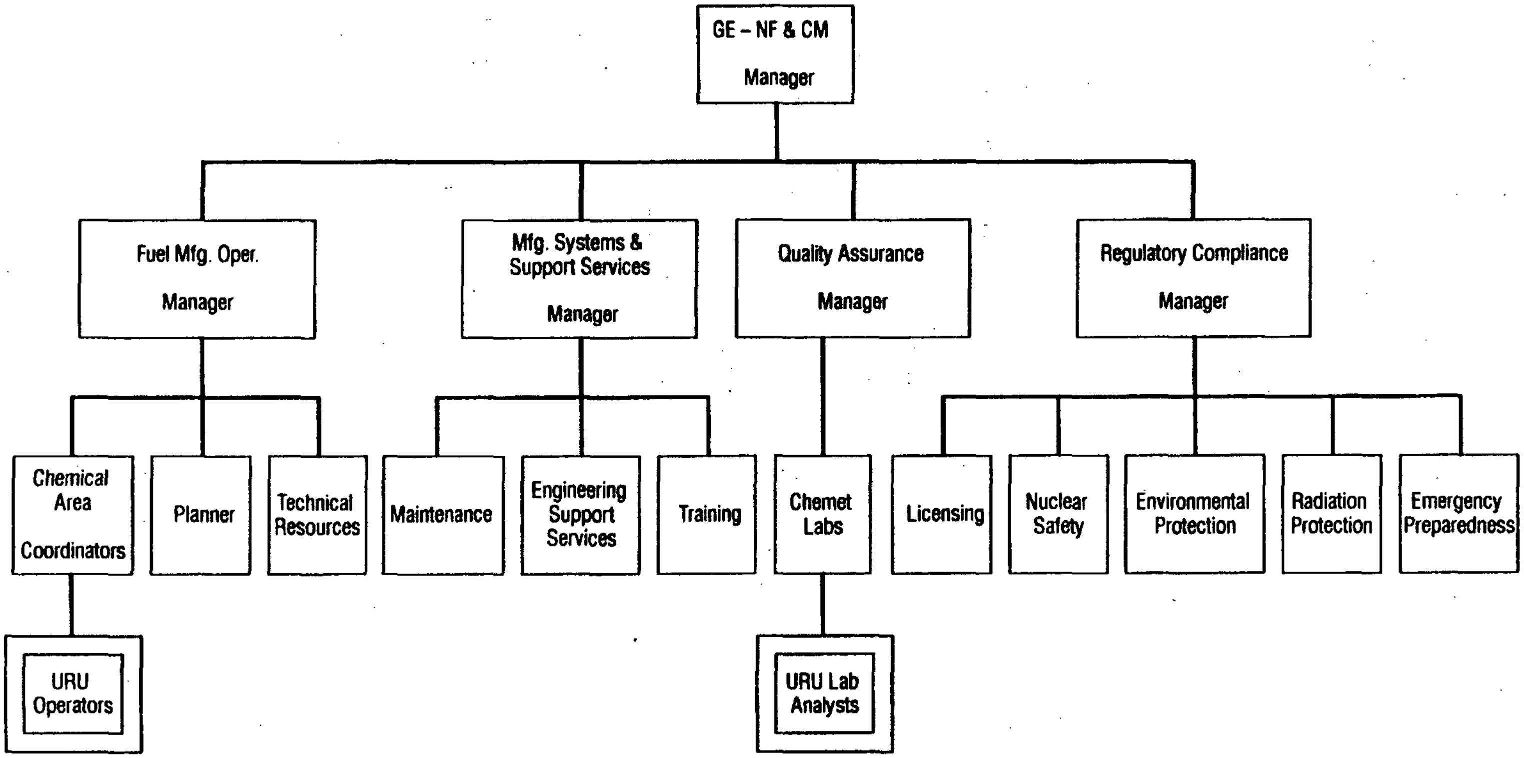


Figure 6.11 Uranium Recycle Unit management oversight and support organizations

# 7 NUCLEAR CRITICALITY SAFETY

## 7.1 Introduction

One of the major objectives in the processing of nuclear fuel with enriched uranium is the avoidance of a nuclear criticality accident. Such an accident would likely result in high levels of neutron and gamma radiation in the immediate vicinity. Such an event in a fuel fabrication facility, which does not normally have much radiation shielding, would be a significant industrial accident. The radiation from a typical accident would likely be lethal to anyone up to 10 feet away in the absence of shielding and would cause exposures of 5 rad at 45 feet (approximately 15 meters).[1] Offsite radiological impacts would not be expected. Equipment damage would not be expected.

To date, eight criticality accidents have been reported in nuclear fuel processing plants (see Table 11.1). All of these involved enriched uranium or plutonium in solution. All involved inadequate procedures and other deficits in management controls. Three of them involved the improper operation of valves and one was caused by the absence of a valve. Two of the accidents caused fatalities and in two others deaths were avoided by prompt evacuations. A nuclear criticality incident occurred naturally in geological history in Africa involving the equivalent of 3.5 percent enriched uranium.

Fuel fabrication facilities, such as GE-Wilmington's, protect against criticality accidents in a variety of ways. The technical basis for controls on uranium mass, enrichment, concentration, and container volume will be presented in Section 7.2. The application of these concepts to specific equipment and operations is determined by the licensee. The margins of safety of specific equipment are evaluated by criticality safety specialists. These specialists frequently use computer codes to model the facility equipment to determine controls and limits necessary to ensure safety. These controls and limits are implemented by equipment, such as valves and monitors, as well as by administrative procedures. The operability of safety-related equipment needs to be ensured by proper maintenance. Compliance with operating procedures needs to be ensured by facility supervisors and periodical audits.

This section will survey the general problem of criticality control in large tanks, evaluate the licensee's criticality safety program, review the licensing and criticality safety issues related to the large tanks of this incident, discuss the potential for a criticality event in V-104, and conclude with a discussion of recent related events at this facility. Other sections of this report address supervision and maintenance.

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[1] Typical in this context would be an energy release of $10^{17}$ fissions.

## 7.2 Criticality Safety of Large Tanks

Two major factors that influence the likelihood of a criticality accident in a large tank are the mass of uranium and the volume of the container. A recent compilation of data[2] indicated that 36 kgs of uranium enriched to 5 percent (U-235) is the minimum amount necessary to cause criticality; for uranium enriched to 3.2 percent, 78 kgs are necessary to reach critical mass. The industry standard for nuclear criticality safety (ANSI/ANS 8.1, 1983) indicates that 32.8 kgs of 5-percent enriched uranium is the subcritical limit value. This value accommodates the uncertainties in the experiments and calculations underlying the data. The relative relationship between critical mass and volume for several enrichments and concentrations is illustrated in Figure 7.1. The percentage of uranium enrichment is shown for each curve. Controls on mass, volume, enrichment and concentration are used in some fashion at the licensee's facility.

## 7.3 Nuclear Criticality Safety Program

The GE license (SNM-1097) specifies that the manager of this site has the responsibility for criticality safety at the facility. This responsibility is delegated to the Area Manager for Fuel Manufacturing Operations. The responsibility to provide authoritative advice to the control of nuclear criticality is assigned to the Nuclear Safety Engineering staff. This staff reports to the manager of Regulatory Compliance, who is assigned the responsibility to establish a criticality safety program to ensure compliance with government regulations.

### 7.3.1 Staffing

The nuclear criticality safety staff at GE-Wilmington consists of two senior specialists, one junior engineer with a full-time assignment in criticality safety, one part time engineer whose major assignment is in radiological engineering, and two criticality safety technicians, who prepare computer code input as specified by others.

The two senior staff members have a substantial amount of experience analyzing criticality safety issues. The junior engineer has had relevant experience. The strength of this staff is in the physics of nuclear criticality safety and in the identification of controls for identified criticality safety risks. The staff's vulnerability is in the knowledge of process engineering and the possible consequences and risk of process upsets or inappropriate operations. The staff is particularly strong for a low-enriched uranium processing facility.

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[2] H. C. Paxton and N. L. Pruvost, Critical Dimensions of Systems Containing $^{235}$U, $^{239}$Pu, and $^{233}$U, 1986 Revision, LA-10860-MS, July 1987.

### 7.3.2 Methods

The licensee uses a variety of methods to determine specific criticality safety limits and margins of safety. These methods are well suited to provide the needed information. The computer codes used are consistently documented and validated for their application at this facility. The codes are capable of three-dimensional modeling of operating equipment and for locating the margins of safety for the conditions modelled. These codes make use of libraries of detailed physics data on the materials of interest.

### 7.3.3 Nuclear Criticality Safety Reviews

The Nuclear Safety Engineering group reviews Facility Change Requests (FCR). The licensee control document for this activity is Practices and Procedures document (P/P) 40-05, "Nuclear Safety Review System." Section 2.1 of P/P 40-50 requires that:

> Components [sic] which initiate additions or changes that may affect Nuclear Safety in Fuel Manufacturing facilities, or other uranium handling facilities are responsible for initiating and completing a Process and Equipment/Facilities Change Request for new or modified processes, equipment or facilities in their areas of responsibilities to assess the need for a nuclear safety review.

Section 2.2 of P/P 40-05 requires that component managers "assure that the need is evaluated for an FCR for proposed equipment repair/replacement activities." Section 2.5 further specifies that the "Area Manager is responsible for evaluating new activities or changes in activities to determine whether a nuclear safety review is required and assuring that required reviews are completed prior to implementation of the change." Factors for the Area Manager to consider in determining if a change may affect nuclear safety are specified in Section 4.2.3 of P/P 40-05. They focus on changes in equipment dimensions, structural integrity, and the control systems.

If the change is not designated as warranting an FCR, it will not get reviewed by Nuclear Safety Engineering. During day shift operations, the planners and process engineers would normally be involved in deciding if the change requires an FCR. During the other shifts or weekends, the Area Coordinators decide the change designation. Managers interviewed by the team reported that they would not normally challenge these staff decisions. Those interviewed indicated that the threshold for designating a change requiring an FCR, with the potential for nuclear safety review, is relatively high. The planners and operators consider the FCR system to be reserved for major facility modifications or component additions, yet they have a narrow view in that they consider only the details of the change and not its effect on the larger system. The licensee does not provide routine, scheduled, and formal training on Nuclear Safety Review System procedures or on the criteria for making these decisions. Audits were not evident to uncover and rectify improper decisions on whether to implement the FCR process.

The team concluded that management control and training for the FCR process were not effective. Examples of weaknesses in this process are discussed in Section 7.5.1 and in Sections 6.1.2, 8.1.4, 9.2.3, and 10.1.

### 7.3.4 Safety Analysis Process

If a criticality safety analysis package is initiated by an FCR, a Technical Report is normally developed by the sponsoring organization, which then becomes the basis for a Nuclear Safety Analysis (NSA) prepared by Nuclear Safety Engineering. The GE-Wilmington P/P document 40-04 "Nuclear Safety Design Criteria," specifies that this Technical Report "contain the information necessary to allow a criticality and/or radiological evaluation" and that it address "1) Changes in the location, containment, or spacing of fissile material, 2) The introduction of hydrogenous materials (water, oil) into moderation control areas, and 3) The failure of passive, active, or procedural criticality safety controls." The intent of the report is further detailed in Section 4.2.4 and Appendix A of P/P 40-05, Rev 6, which emphasizes credible accident conditions and proposed control methods and equipment.

When the Technical Report is accepted by the Regulatory Compliance organization, the NSA is developed. The NSA contains the details of the calculations made, modeling assumptions, and the limits and controls necessary to provide adequate safety for the operation. The more significant controls will appear on the Nuclear Safety Release/Requirements (NSR/R) sheet issued with the NSA, and which is intended to be part of the operating procedures. The NSA is considered an internal document for the Regulatory Compliance organization and is written only for the files.

The Nuclear Safety Engineering staff specifies the process controls (NSRs) sufficient to maintain acceptable margins of safety and provides contingencies against the variety of paths which could lead to a criticality accident. The operating or process engineering organizations do not document agreements that the range of upset conditions considered are comprehensive or that the specified NSRs are adequate to control the risk.

### 7.3.5 Nuclear Criticality Safety Audits

The criticality safety audit program is required by the GE-Wilmington license (SMN-1097), Section 2.8.1, and is described in the licensee's P/P 40-06 and NSI-E-2.0. The license requires that a quarterly audit of nuclear manufacturing and support areas be conducted by senior criticality safety staff and that "Such audits are performed to determine that actual operations conform to criticality and radiation safety requirements." The focus of the implementing procedure, NSI-E-2.0, is on regulatory compliance. A compliance audit is conducted to ensure that attention is paid to known risks which have a regulatory commitment. A "search out" audit or appraisal is used at some facilities to detect problems not yet known. Search out audits are not formally assigned at this facility.

Team interviews established that the focus of licensee nuclear safety audits was equipment and configuration changes which would affect the Nuclear Safety Analysis. Process operations and the implementation of controls are not normally a focus of these audits; indeed, the background of the auditors makes it unlikely that they would focus on process operations. These are not compliance audits. Even if the intent was to review compliance with nuclear safety limits, the licensee's record keeping of such compliance would make this task very difficult, if not impossible.

The team concluded that the licensee did not have an effective operational and safety audit program. An example of such an audit deficiency is discussed in Section 7.5.2. It is clear that the licensee's audit program did not uncover and cause corrective action on the pervasive breakdown of criticality controls discussed later in this section.

### 7.3.6 Nuclear Criticality Safety Training

New employees to the GE-Wilmington site receive a half day orientation to safety issues. About 30 minutes of this presentation is devoted to nuclear criticality safety and covers the concept of critical masses and chain reactions, criticality control methods generally used, and provides examples of specific controls, followed by a review of one historic accident. Although the training notes that a criticality accident could lead to fatalities among nearby personnel and the likelihood of plant closure, it also mentions several times that a criticality accident has never occurred in a low-enriched plant, such as GE-Wilmington's, and that the nuclear safety evaluations for the plant are very conservative. A test on the presentation is given and a passing grade is required for unrestricted plant access. An annual retraining lecture on criticality is required for continuing site access. The most recent lecture lasted 45 minutes and reviewed basic concepts in criticality safety and emphasized known problem areas in the plant.

This formal training is uniform for all employees and does not vary by job assignment. On-the-job training by supervisors and senior operators is the basic method for learning specific criticality controls for specific operations. However, those responsible for providing the on-the-job training do not receive additional instruction on the criticality safety issues for specific operations. The Nuclear Safety Analysis for specific operations normally covers the need for criticality requirements and limits, the assumptions on which the controls are based, and the consequences of limit violations, but they are not used as a training aid.

The team concluded that licensee training in criticality safety met minimum regulatory commitments. However, it did not properly emphasize the importance of compliance with the specific criticality control requirements.

## 7.4 Licensing the Uranium Recycle Unit (URU)

### 7.4.1 Criticality Safety Principles in the URU License Application

On December 3, 1984, GE sent the NRC an application to amend license SNM-1097 to include the Uranium Process Management Project (UPMP). (This system is now called Uranium Recycle Unit (URU)). On February 26, 1985, the NRC approved the application. Chapter 4 of the application detailed the nuclear criticality control principles. These principles are consistent with those in license SNM-1097. The basic principle is that

> Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible.

The application also stipulated in Section 4.1.4 that "where geometric control is not practical, criticality control may be based on control of U-235 mass or control of moderation." Another possible component of criticality control is the chemical form of the uranium. These issues are addressed in Section 4.1.7: "In criticality safety analyses, the optimum credible form of fissile material and moderation is assumed to be full density $UO_2$ and water mixtures." In lieu of developing an argument for a less conservative compound or establishing chemical controls to assure other chemistry, $UO_2$ and water is assumed to be the optional chemical form for the basis of safety.

### 7.4.2 Proposed Specific Controls

The GE-Wilmington operations of interest in this incident are shown in broad scope in Figures 4.3 and 4.4. Solvent extraction (SX) and waste processing are detailed in Figures 3.3 and 3.4, respectively. Uranium from several sources is prepared in feed adjustment tanks and fed to the SX system. As seen in Figure 3.3, the SX waste stream leaves the bottom of Column A and is accumulated in favorable geometry quarantine tanks V-290 and V-291. Figure 3.4 shows that the quarantine tanks can be discharged to a rework tank (V-225) or to the unfavorable geometry waste accumulation tank (V-103).

Chapter 2 of the URU application describes the specific controls for the various process operations within the facility. Table 2.4-10 summarized the eight operations of the Scrap Processing Operation and the two or three criticality safety controls for each operation. Section 2.4.3.2 provided the details of the controls, including the controls for systems of primary interest in this incident, the SX system and waste tanks. As shown in Figure 3.4 of this report, the nitrate waste system also feeds tank V-103; this system and associated controls are described in Section 2.3.2 of the application. The criticality safety of the 20,000-gallon tank V-103 is based on controlling three parameters. Section 2.4.3.2.7 presents a system of controls and Section 2.4.3.4.3 and Table 2.4-10 presents another control system, but the three control methods evident are:

1. Process control
2. Mass limit control
3. Density limit control

<u>Process control</u> would be implemented by a digital computer process control system which would automatically control the process parameters in such a way as to preclude unsafe quantities of uranium from leaving in the waste stream.

<u>Mass limit control</u> would be implemented by (1) near real-time sampling and analysis by an automatic system of the waste stream in T-292 prior to discharge to the quarantine tanks, and (2) analyzing the uranium content of quarantine tanks before release to tank V-103. The mass limit of 12 kgs of uranium was selected as a safe batch considering uncertainties of the uranium heel in V-103. The mass limit in a full 20,000-gallon tank resulted in the 150 ppm of uranium concentration administrated to assure compliance.

<u>Density limit control</u> would be implemented by (1) providing recirculation to assure a homogeneous mixture of any solution and solids in tank V-103, and (2) a density probe and controller to close inlet valves if the appropriate setpoint is exceeded.

### 7.4.3 NRC Review of the Licensee's Application

The NRC Safety Evaluation Report (February 26, 1985) written to support the issuing of the license amendment authorizing the URU notes that certain criticality safety controls will function automatically and are not controlled by the process operator. These automatic controls were routing material from the quarantine tanks on the basis of acceptable uranium concentrations and the density probes with alarms and automatic cutoff capability of feed to unfavorable geometry tanks.

The URU was authorized by SNM-1097, Amendment No. 3, dated February 26, 1985. The licensee's December 3, 1984 submittal was incorporated by reference in its entirety into License Condition 9. By incorporating License Condition 9, the NRC reviewer intended to ensure that the conditions and limits of the submittal continued in use by the licensee and that proposed changes would be authorized only by license amendments. In direct contrast, the licensee did not consider the incorporation of the December 3, 1984, submittal into License Condition 9 as limiting their ability to unilaterally change the URU process and controls under the administrative provisions of their license. The NRC inspections were predicated on the same understanding as the licensee's. It is also notable that while the license authorizes uranium enriched to six percent, the safety basis for URU was demonstrated for uranium enriched to five percent.

The team found that the licensee and the NRC did not mutually understand that incorporating a license condition by reference limited the licensee's authority to make changes. (See Section 12 for additional details.)

## 7.5 Implementation of Controls for Tank V-103

Implementation of mass and density limit controls on tank V-103 were subsequently defined in GE Technical Report 3.13.2, issued on March 27, 1985. The report listed two Active Engineered Control (AEC) systems to support criticality safety. The dual sampling from recirculation tank T-292 and from the quarantine tanks was the first AEC to provide protection from high uranium levels in SX waste. The second AEC was the system to ensure recirculation in tank V-103 with two primary pumps augmented by a backup pump. Included in the second AEC are two density probes with an automatic input shutoff setpoint capability. The SX process and its control system are not presented as a contingency, but were installed as described in the application. This control system was consistent with the assumptions of the NRC Safety Evaluation Report. Since that time, however, the anticipated controls have eroded, as discussed in the following section.

### 7.5.1 Established Controls

**Mass Limit Contingency**

Two automated sampling systems originally were used to administer the mass limit. The first system was a continuous flow-sampling line from recirculation tank T-292 to the laboratory for periodic measurements. The second system was by a recirculation sample line to the laboratory from quarantine tanks V-290 and V-291. By 1987, the first circulating sampling system from T-292 was no longer used and was replaced with occasional manual sampling from a tap on T-292 or on the A column. The manual sampling apparently occurred if the quarantine tank sample prevented a transfer to V-103 and the operators needed a second opinion. As a result of deleting the T-292 automatic sampling system, the remaining sampling measurements had a much higher vulnerability to error because of the lack of redundancy.

**Circulation Contingency**

The second contingency was provided by a requirement for constant circulation by pump of the contents of tank V-103. This requirement was added to reduce the criticality risk from the settling of suspended uranium solids into an unsafe density, on the assumption that a well-mixed uranium slurry would be safe. The tank is equipped with two primary pumps connected in parallel and with a backup pump, connected to emergency power, with a separate circulation loop. This configuration has the potential of being a well-defended barrier if the mass control fails.

The feed to the SX system in the URU is from tanks with favorable geometry in the feed preparation system, which in turn, is fed from uranium scrap dissolution operations. Until 1987, the feed tanks to the SX system were monitored by density instruments which were set to alarm at a uranium density of 280 grams per liter (gU/l). This density would not cause a criticality risk for five-percent enriched uranium in large downstream tanks, such as V-103 or V-104, as long as the uranium remained in solution or was effectively mixed.

If the SX system is not effective in extracting uranium, the waste stream (AW) will exit with essentially the same concentration as the feed stream. In 1986, a facility change request was made (FCR 86.226) that the requirement for these density monitors be deleted. Nuclear Safety Engineering did not agree with the request, as written, but agreed to the compromise of raising the alarm setpoint for these monitors to 350 gU/l. This concentration was determined not to be a risk in the feed preparation system, but downstream systems do not appear to have been considered.

Uranium concentrations at the higher feed tank set point would not be critically safe in a tank as large as V-103. The set point change compromised the fundamental purpose of the V-103 circulation contingency. A safety requirement could have been developed, although it was not, for a mandatory non-uranium heel of sufficient volume to accommodate the circulation requirement.

**Density Probe Contingency**

The third contingency was originally provided by a density monitor for tank V-103 with a set point alarm and automatic closure of a block and bleed valve. This density probe would presumably detect any unsafe precipitation of uranium or unsafe solutions which sank to the probe level. By 1987, this control had been deleted through FCR 87-125. The Criticality Safety Analysis of June 4, 1987, approving this change, concluded that three controls were not necessary because the remaining two controls met license requirements.

Of the three layers of contingency controls presented in the December 3, 1984, application, and reviewed in subsection 7.4.2 of this report, the first (process control) was installed but not subsequently considered to be a criticality safety contingency barrier. The second was significantly weakened by reliance on a single sampling system. The third contingency control was partially deleted. In summary, for uranium solutions not inherently safe by concentrations, only the sampling barrier remained.

The team concluded that subsystem changes to processes do not receive adequate analysis for their potential effects on the safety of the whole system. A corollary finding is that the licensee lacked a multidisciplinary approach to identify (1) each route to a criticality scenario, (2) all contingencies for each scenario, and (3) formalize effective controls for each contingency.

### 7.5.2 Assurance of Remaining Criticality Barriers

**Assurance of Sampling**

Because it was the only remaining barrier for some criticality scenarios, the sampling process system for the quarantine tanks increased in importance. It was essential that this process work well. However:

- The design of the circulation sampling system caused it to be vulnerable to not finding undissolved uranium particles, precipitated uranium particles, or uranium in the organic phase. The systems were also prone to plugging. Numerous such problems with non-representative sampling systems at the URU had become evident through precursor events (see Section 11).

- The automatic feature of the circulating sampling system proved troublesome and the laboratory technicians lost confidence in it. Manual sampling taken in the laboratory gradually replaced automatic sampling. In this mode, the sample results were verbally reported to the SX operators so they could enter the results in the computer system. The possibility of data transmission errors that would degrade the intended reliability of the overall control was always present (see Section 6.1.2).

- Another degradation of this contingency was the capability and standard practice of bypassing the requirements of the computer control system. Compliance with the release limit and the requirement to sample before release to the quarantine tanks were defeated when operators ran the process in TUNE mode. (See Section 8.1.2 for an explanation of TUNE mode operations.) The record keeping of sampling and transfers from the quarantine tanks made it extraordinarily difficult to determine if operators complied with requirements.

Further, the team could find no evidence of management audits of the sampling process. If management does not consider audits important and if noncompliance was hard to uncover, it would be unreasonable to expect consistent application of the Nuclear Safety Requirements for discharges from the quarantine tanks.

The team concluded that the mass contingency had degraded because of management's inattention to the point where it was no longer a valid barrier. (See Section 10 for additional details.)

**Assurance of Circulation**

The circulation contingency is defended by three pumps at the tank. The effectiveness of these pumps in providing the required service depends on the correct alignment of a number of valves. These valves have been designated "Critical Valves" and are so listed on NSR/R 04.05.03. The listing identifies the valves by function, because they do not have unique identification. During facility tours, the team noted that it took system experts some time to identify which valves corresponded to the valves listed the NSR/R. Many, but not all, of the critical valves have a red "Critical Valve" plaque near them. The plaques do not have unique identifiers nor the correct valve positions.

The team concluded that an unambiguous listing of equipment important to safety and a clear identification of equipment would provide stronger assurance that correct action will be taken.

### 7.5.3 Findings

The system of controls for criticality safety in tank V-103 was originally extensive and provided true defense-in-depth. The circulation control in the tank could be considered an after-the-fact action and thus not a true contingency, but it provides real protection nevertheless. Control system features were disabled or deteriorated as unit operations proceeded until they consisted of only one workable barrier to criticality, the sparging capability, and even it did not protect against all criticality scenarios. At the time of this incident, the uranium mass limit contingency barrier had degraded to the point where it was no longer adequate.

## 7.6 Criticality Safety of Tank V-104

The contents of tank V-103 are batch transferred to another 20,000-gallon tank, V-104, located about a quarter mile away at the Waste Treatment Facility (WTF). Lime is then added to tank V-104 to precipitate the uranium and other heavy metals, which settle out as solids and are removed by circulation through a filter.

**Established Criticality Controls**

Criticality safety control of this vessel is analogous to that for tank V-103. The Criticality Safety Analysis (CSA) for FCR 84.054 (dated March 26, 1986), identified that the first contingency in batch mode operation (current licensee practice) is by the mass limit of 35 kgs of uranium. This is 90 percent of the critical mass, assuming uranium dioxide enriched to five percent and mixed with water. From this limit mass and the volume of the tank, a concentration limit of 500 ppm uranium for incoming streams was determined to verify the limit. The second contingency identified was density control. The density monitor had two actions limits at 1.1 g/cc it would initiate operation of the filter and at 1.3 g/cc it would initiate air sparging and shut off feed to the tank. The team notes that this density control mitigates the results of an upset condition. A better contingency would guard against the upset condition occurring in the first place.

By the time of Revision 4 (April 9, 1990) to NSR/R (04.04.05), the density contingency for batch mode operation was to require air sparging during transfers from tank V-103 until tank contents are less than 500 ppm of uranium; then the sparge can be stopped.

A grab sample tap on tank V-104 allows a sample to be taken after a transfer from tank V-103 is complete.

Because tank V-103 is not sampled before the transfer, compliance with the transfer concentration limit relies on the quarantine tank sampling system. The operating assumption is that the 500 ppm of uranium limit would be met since the V-103 limit is 150 ppm. Problems with the sampling are discussed in Subsection 7.5.2 above.

**Safety Margin in Tank V-104 During the Incident**

At the time of the incident, the SX system was processing 275 kgs of uranium. Because of process upsets, 150 kgs of this material were transferred to the waste system and to tank V-104, in particular. The safety limit of 35 kgs was exceeded by a factor of 4. If the 275 kgs available had been transferred, the limit would have been exceeded by a factor of about 8.

In this incident, the criticality accident was prevented both by the sparging and the fortuitous presence of large quantities of other solids. The sparge was the remaining safety control; the waste solid quantity was not. The safety analysis and controls for the criticality safety of tanks V-103 and V-104 appropriately do not consider chemical forms of uranium less conservative than $UO_2$ and water. Processes with a variety of possible chemicals feed into this system, so controlling the form of uranium and the effect of diluents would be a difficult basis for a control. The diluents would vary in amount and would likely have different settling rates than heavy uranium materials. The presence of solid waste materials, in this case to dilute the solid uranium, meant that the criticality safety of the tank was then not wholly dependant on the effective functioning of the mechanical equipment making up the sparge system. The amount of non-uranium solids in the tank is not controlled, however, and could not be considered a basis for safety for future operation of this system without a significant revamping of the safety controls and supporting analysis.

The V-104 Criticality Safety Analysis (March 26, 1986) listed the neutron multiplication ($k_{inf}$) values for selected uranium compounds at various mixture densities. The critical condition occurs at $k_{inf}$ = 1.0, which is the failure to be avoided. ($K_{inf}$ is the neutron multiplication factor for fissile material in very large volumes, such as tanks V-103 and V-104.) Figure 7.2 shows mixture density versus $k_{inf}$ for uranium enriched to five percent. The operationally expected maximum enrichment of 4.025 percent is illustrated in Figure 7.3 and indicates that 1.37 g/cc and 1.45 g/cc are critical mixture densities for uranium oxide and calcium uranate, respectively, in a mixture with water.

The possible densities of settled solids in tank V-104 are a significant safety issue. These are determined by the crystalline density of the solid and the void fraction in the settled state. One study reported that the void fraction in settled sludge varied from 70 to 90 percent[3]. This study covered all forms of precipitated waste from nuclear processing. Data from the

[3] R. F. Bradley, et al., "A Low-pressure Hydraulic Technique for Slurrying Radioactive Sludges in Waste Tanks," DP-1468, E. I. du Pont de Nemours and Company, November 1977.

Idaho National Engineering Laboratory on settling characteristics of undissolved fuel from waste recovery operations determined that the initial void fraction was 85 percent for dried solids and 75 percent when the solids were agitated. Wet solids had a slightly higher void fraction.

Section 15.4.1.3 of the current license (SNM-1097) specifies that the crystalline density of $UO_2$ is 10.96 g/cc, while the maximum bulk density had been determined experimentally to be 4.5 g/cc. This means that the minimum void fraction (VF) for the bulk form of $UO_2$ is 59 percent. Section 4.2.7.2 of the license specifies that "A full density mixture is used in determination of uranium concentration (i.e., the effects of voids or inert materials mixed with the accumulation is not included)." From Section 7.4.1 of this report, $UO_2$ and water are specified as the chemical forms to establish safety in the URU. Although tank V-104 is downstream of the URU, the same principle should apply.

From published crystalline densities, and considering that the range of void fraction (VF) from 0.7 to 0.9 needs to be considered, the anticipated mixture densities for selected uranium compounds are given below:

| Compound | Mixture Densities in g/cc | | | |
|---|---|---|---|---|
| | crystal | VF=0.7 | VF=0.8 | VF=0.9 |
| $UO_2$ | 10.96 | 3.99 | 2.99 | 1.996 |
| ADU | 8.39 | 3.217 | 2.478 | 1.739 |
| $Na_2U_2O_7$ | 6.5 | 2.65 | 2.10 | 1.55 |
| $Ca_2U_2O_7$ | 5.6 | 2.38 | 1.92 | 1.46 |
| $H_2UO_4$ | 5.93 | 2.48 | 1.99 | 1.49 |
| $UF_4$ | 6.7 | 2.71 | 2.14 | 1.57 |
| $UH_3$ | 10.95 | 3.99 | 2.99 | 1.995 |

The range of mixture densities in the table for $UO_2$ and $Ca_2U_2O$ are larger than the densities for $K_{inf}$ = 1.0 shown in Figures 7.2 and 7.3. $Ca_2U_2O_7$ is the more likely solid, but $UO_2$ was properly chosen as the conservative case. However, most enriched uranium compounds precipitating and forming a settled sludge without diluents could result in densities which are not critically safe. This is the reason for the sparging requirement for tank V-104.

As part of uranium recovery actions, the licensee performed criticality safety calculations for the specific geometry of the lower cone of tank V-104. These calculations showed the minimum critical mass to be about 100 kgs for the 3.2 percent enrichment of the batch processed. They also showed that a mixture density above 1.6 would be required if just

150 kgs of uranium were transferred to tank V-104 to cause a criticality accident. The team concurs with these results.

A study of the settling characteristics of lagoon sludge at GE-Wilmington[4] found densities in the range of 1.04 to 1.12. A previous report[5] found sludge mixture densities in V-104 up to 1.077. These lighter waste materials are expected to accumulate in this waste tank. These diluent materials, if mixed with the heavier uranium compounds, would lower the densities from that shown in the above table for single components.

At the time of the V-104 incident, the density of a sampling from the solids that had been centrifuged as part of the recovery action was 1.269. The uranium content was less than 20 percent by weight. The centrifuged material density presented a best-case scenario since the vigorous sparging assured that the centrifuged solids were well mixed. The heavier uranium solids would be expected to settle out before the other solids in the tank without sparging. The density in the heavier stratification layer would have been significantly higher than that measured if the sparging had stopped, but with the 3.2 percent enrichment and low-percent uranium in the solids, a criticality event was unlikely with only 150 kgs.

If the inventory of the 275 kgs available had been transferred to tank V-104, the well mixed density should have exceeded 1.4 g/cc and the uranium layer of a postulated unsparged sludge would be significantly more dense than this. In this case, the sparging would likely be necessary to prevent the criticality accident. Fewer non-uranium diluents would make the situation even worse. Without the presence of the non-uranium solids, a criticality event could not be precluded without the effectiveness action of the sparge.

The team concluded that as this waste system is currently designed and operated, a nuclear criticality is a viable accident scenario that must continue to be prevented with effective controls.

## 7.7 Criticality Safety Implications of Recent Events

### 7.7.1 Release from Radwaste on September 18, 1990

On September 18, 1990, uranium in the range of 30 to 300 kgs was released from the Radwaste system and eventually discharged to the site aeration system. This uranium was solid and would normally have been contained within the bank of filters used to separate solids from liquids. In this case, the filter media failed and uranium solids flowed first to a favorable geometry tank (T-630) and then were pumped to an unfavorable geometry tank (T-632). Both of these tanks were continuously discharged. The nuclear criticality control requirement

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[4] Gregory D. Garstka, "Settling Characteristics of West Nitrate Lagoon Sludges," December 28, 1989.

[5] R.A. Oler, "Sludge Characterization Study: V-104, Interim Report," 76PTM046, April 30, 1976.

was that the discharge from T-630 be sampled through a line leading to the URU laboratory in a period not to exceed two hours. T-632 discharged to an unfavorable geometry pH-adjustment tank which, in turn, fed the aeration basin. The licensee's best estimate is that 145 kgs were released.

The basic criticality safety control was tank T-630 discharge sampling with a concentration limit of 25 ppm uranium. To guard against a large uranium release, for which the two-hour sampling period would not be adequate, the licensee relied on a differential pressure monitor across the filters. After this incident, the licensee realized that the differential pressure monitor was ill conceived and would not signal a problem. It was also seen that the presence of solid uranium in the discharge line would likely plug the sample line and leave no timely barriers to prevent large amounts of uranium from getting to unfavorable geometry tanks or basins. These large vessels were "pass through" devices on the assumption that uranium containing solids would not settle until reaching the basin.

The licensee's incident report identified a short-term fix of authorizing batch releases from tank T-630 after manual sampling. The proposed long-term fix was to install a uranium monitor for the line. A separate root cause analysis by the licensee identified problems with personnel, equipment, procedures, and systems. The personnel section highlighted difficulties with management control, communications, and worker attitude. Problems areas for equipment were design, installation, maintenance, and knowledge base. System problems were attributable to inadequate functional testing, while procedures were seen as lacking completeness. The team considers this a perceptive evaluation and notes many parallels with the incident considered in this report. The team also notes that a comprehensive safety analysis would have identified the need for requirements in the areas of most of these weaknesses and that these requirements could have generated programs. The September 18, 1990, incident did not result in such follow through.

### 7.7.2 Release from Fluoride Waste System on February 5, 1991

On February 5, 1991, some 33 kgs of uranium were released from the fluoride waste system into the unfavorable geometry tank V-106.

During normal operation of the fluoride waste system, uranium collects on the ion-exchange resins which half fill each of three favorable-geometry 1415-liter columns. The nominal loading rate of this resin material is 50 gU/l, with a possible loading rate of some 93 gU/l. The resin material is "regenerated" to remove the uranium in a three-step operation. In the first step, the fluoride waste liquid in the top half of the column is transferred to tank V-106. In the second step, approximately 700 liters of ammonium bicarbonate (ABC) solution are added to the column and the displaced liquid flows to tank V-106. In the third step, the valving is set to transfer the column contents to favorable geometry tanks; more ABC solution is added to the columns in order to release the uranium from the resin.

In this event, the instrument that controls the amount of ABC flowing to the column was erroneously disabled, allowing 4800 liters of solution to enter the column and into tank V-106; about some 33 kgs of uranium flowed in with it. If two of the three columns are loaded to the high level and are being regenerated concurrently, then the available uranium for a similar upset could exceed 125 kgs. The nominal loadings would provide about 67 kgs. The third column would add proportionally more uranium if connected in this operation, although such a lineup would be unlikely. These amounts are large compared to safe values. Regenerating multiple columns would not be a normal practice, but the barriers to keep critical mass quantities of uranium out of tank V-106 during regeneration of the ion exchange columns appear less than adequate, a condition not addressed in the licensee's incident report. However, since the team's visit to the plant, additional controls have been instituted for this operation.

### 7.7.3 Use of Incidents

The team concluded that the licensee did not fully benefit from the initial investigation of these incidents by using them to strengthen criticality safety analyses and to develop needed controls.

## 7.8 Findings and Conclusions

- A criticality accident did not occur in tank V-104 tank because of sparging operations and the presence of large quantities of non-uranium solids in the tank. As the licensee's waste system is currently operated, a nuclear criticality accident is a viable risk.

- The system of criticality safety controls for tank V-103 was originally extensive and afforded true defense-in-depth. However, this system of controls deteriorated as unit operations proceeded. The process controls and the mass limit control were no longer effective, with the recirculation capability the only workable barrier against a criticality accident. However, the recirculation barrier would not protect against all postulated scenarios.

- In reviewing the licensee's nuclear safety program, the team determined that the criticality safety staff was unusually strong for a low-enriched processing facility. However, the licensee lacked an effective multidisciplinary approach to take advantage of this staff's strengths to identify (1) each route to a criticality accident scenario, (2) necessary contingencies for each scenario, (3) the formal controls necessary for each contingency.

- The team determined that licensee management inadequately controlled the Facility Change Request system. As a result, changes to subsystems did not receive adequate analysis for their potential effect on the safety of the whole system.

- The licensee did not have an unambiguous listing of equipment important to criticality safety nor clear and unique labeling for this equipment.

- The licensee did not fully benefit from the investigation of incidents at this facility by using them to uncover unanalyzed routes to a criticality accident or to develop needed controls.

- The NRC and the licensee did not mutually understand that incorporating a submittal by reference as a license condition limited the facility's authority to make changes.

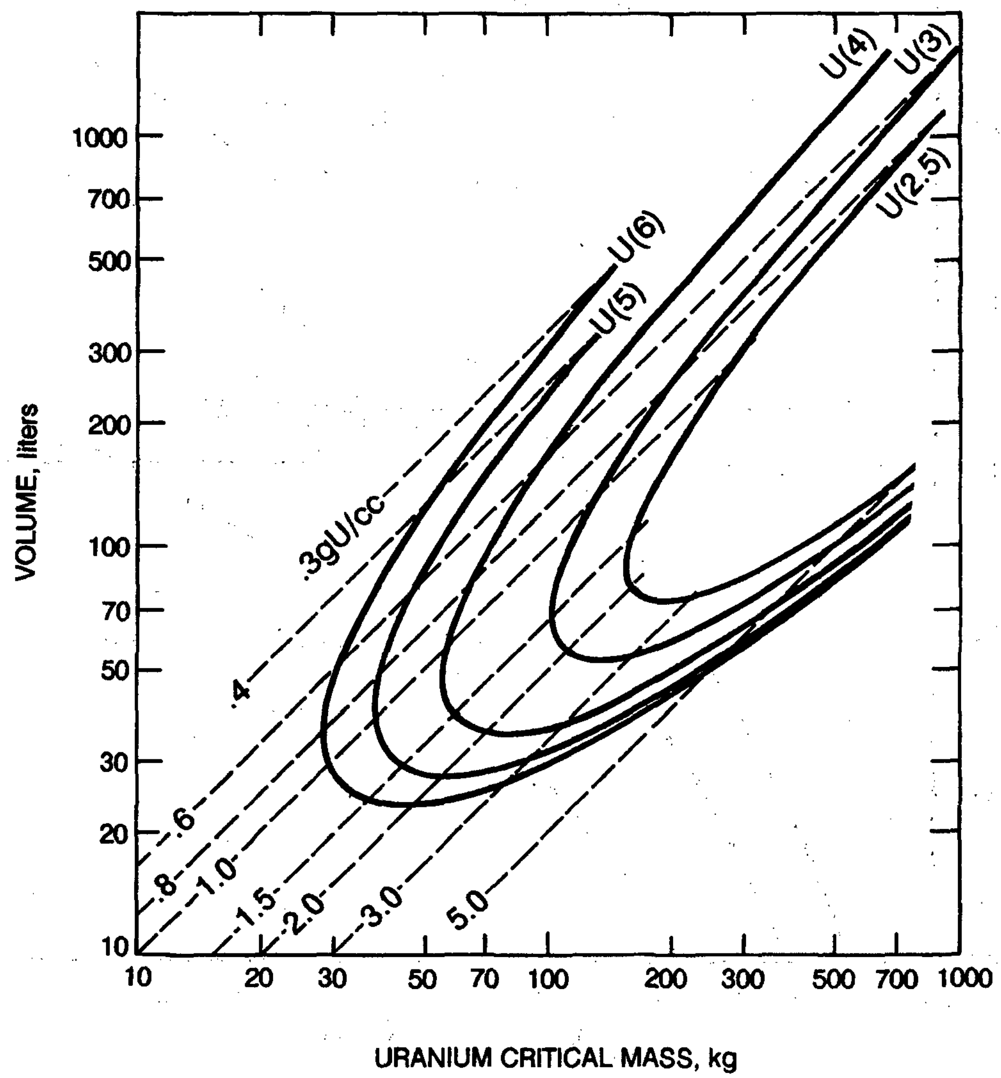



Figure 7.1 Critical Sphere Volume vs. Critical Spherical Mass $UO_2 - H_2O$

Source: R. D. Carter, G. R. Kiel, and K. R. Ridgway, Criticality Handbook, ARH-600, Figure III. B. 9-6, Atlantic Richfield Hanford Company, June 30, 1968.

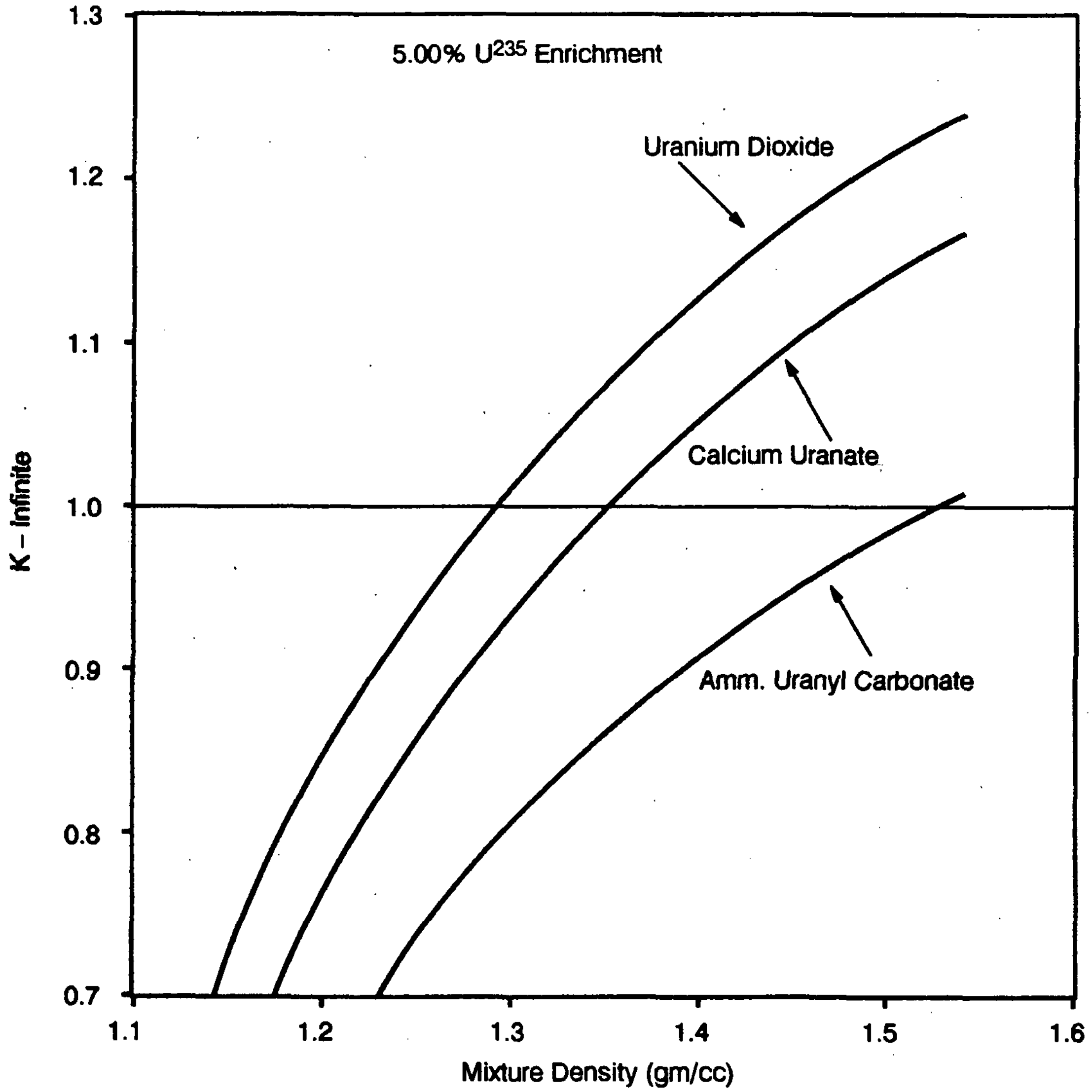



Figure 7.2 K-Infinite as a function of mixture density

Source: General Electric Corp.

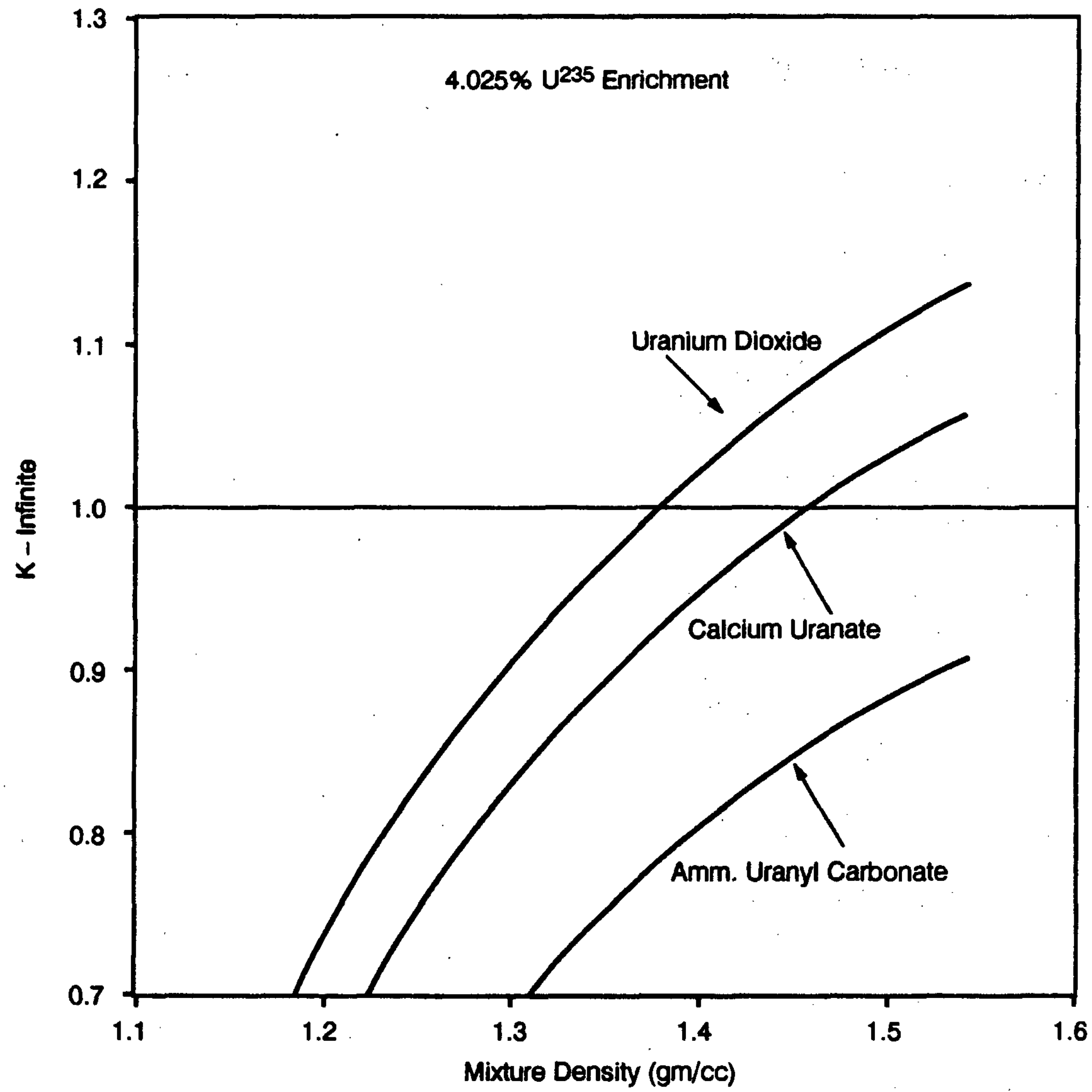



Figure 7.3 K-Infinite as a function of mixture density

Source: General Electric Corp.

# 8 EVALUATION OF INSTRUMENTATION AND CONTROL

## 8.1 Process Control System

### 8.1.1 System Description and Evaluation

The Uranium Recycle Unit (URU) process control system uses commercially available digital computers to automatically control the URU process. The system is a distributed digital control system with both batch and continuous mode processing capabilities with digital communication between the operator consoles and the local control devices. The goal of the control system is to maintain high reliability for processing operations.

The control system employs redundancy only in three areas where high reliability is considered mandatory:

- There are two communication links to assure that the system communicates at all times among the consoles, field inputs, and the local controls units.

- There are dual operator consoles to allow operators to control URU processes from either console and to provide for backup operator interface for all processes requiring high reliability. Key switch operation is used to prevent the simultaneous operation of process controls from both consoles.

- Unit operation controllers are used in a redundant configuration to provide high reliability for those control functions available on both operator consoles. When used in a redundant configuration, one controller provides the primary control function while the other is in a standby state, which is continually updated with current operating data. The backup controller will be ready to assume control if the primary controller fails. Failures will be detected by both internal and external cross-checking features.

Signals to and from field devices interact with the control system. The system has a dedicated microprocessor with its own dedicated sequence programs to operate and control processing units. Control logic sequencing takes place in the unit operation controllers or individual loop controllers. The communications link is used to transmit start-stop commands and to communicate status and process information to the operator's console for interactive graphic display, trending, logging, and operator interaction.

The system gathers process information from the controllers and forms it into a data base suitable for transmission over the data highway and for use by other data highway devices.

Through interviews with operators, the team found that the digital computer control system for URU is user friendly and reasonably reliable from the operator's point of view. There is no evidence that the digital computer control system hardware failed or that software malfunctioned during this incident. However, the team was not able to review the original computer software design information and its qualification documentation (software verification and validation process). The licensee did not maintain the software verification and validation documents. It is the team's understanding that the licensee's engineering staff only performs configuration control management. The original software verification and validation were performed by the computer system vendor. The licensee has a contract with the vendor to maintain system software. Because it is a commercially available computer control system, the licensee is able to obtain upgraded system software to keep up with state-of-the-art computer software development. All the computer system software modifications are performed using the configuration programs in the system. However, the licensee does not perform software verification and validation on the modified software and there is no software quality assurance program for software changes.

### 8.1.2 Manual Override Capability

Two operator consoles, each with two-color cathode ray tubes (CRTs), provide real-time interactive graphics to display all URU process parameters and provide the operator with the means to operate all processes and equipment, respond to alarm conditions, input requested data, request CRT copies and logs, etc. Each console has a four-position key switch:

| Key Switch Position | Function |
|---|---|
| 1. LOCKED | Allows only monitoring of all the displays. |
| 2. OPERATE | Allows all normal operating functions. |
| 3. TUNE | Allows certain process parameters or variable control loop parameters to be changed. |
| 4. CONFIGURE | Allows graphic displays, control sequences, control logic, etc., to be modified or generated through the use of a configuration keyboard on a separate engineering console. |

Through interviews and document reviews, the team found that the licensee does not restrict the operator from entering the TUNE mode (i.e., manual override capability) on the operator console to change process control loop parameters. As originally planned, TUNE mode operation was limited to the shift foreman or the process engineer. Under the original plan, only they were authorized to temporarily bypass or override certain automatic control features. The TUNE mode is not intended to be used routinely by the operator. However, the team found that the operator has ready access of the console mode control key. Under TUNE mode operation, for example, the operator could override the interlock circuitry and open the dump valves to transfer the aqueous waste from the quarantine tank to the unfavorable geometry waste accumulation tank. The operator could also override the interlock circuitry and open the inlet valve at the feed adjustment tank even though the high density limit was reached. Unrestricted use of TUNE mode operation could defeat all the automatic control safeguards. Team interviews with control room operators indicated that TUNE mode operation was the predominant operating mode.

### 8.1.3 Reliability of the Logger and Historical Trend Computer

Each operator's console has one printer and one data logger provided. The logger is capable of alpha-numeric printing only. The printer gives both alpha-numeric printing and black and white copies of CRT graphic displays on operator demand. Batch end reports, hourly, shift and daily logs or summaries, and operator actions and alarms can be printed on either the printer or logger. In addition, there is a digital control system historical trend computer that can be used by process engineers to evaluate process system performance.

The team observed that logger output was very difficult to read and follow. Although it logged some operator actions, there was no management or engineering review effort for tracing operator actions, especially with respect to TUNE mode operations. The logging records were not retained for historical purposes.

Through interviews, the team found that the digital control system historical trend computer was not reliable because it was frequently inoperable. That system was maintained by an outside contractor. However, there is a backup system which stores all the critical historical data off-line. During the period covered by this incident, the trend computer system was down. Essential data for re-construction of the incident had to be obtained from a backup off-line computer data system.

### 8.1.4 Configuration Control

All control system software changes to devices and loop controllers are programmed through a complex process called "configuration." Each type of device has a unique set of configuration programs and any software changes are made via these programs. The configuration program requires responses to a detailed series of questions. Once all the questions are answered correctly, a download file is generated and transmitted over the data highway to the memory of the process control device. Device configurations (program changes) can only be accessed when the console key-lock is in the CONFIGURE position, and the appropriate device configuration is transferred to the configuration logic unit. These changes are performed by the control engineer at the engineer's console.

GE Technical Report 3.13.18 "Process Control System Configuration Control," specifies the configuration control procedure for modification of any "criticality control instruction." It specifies that a Facility Change Request (FCR) must be issued for any criticality-control-related change. The FCR contains a description of the change and why the change is required (i.e., its basis). Approval of the change is made in accordance with FCR Practices and Procedures No. 40-5 "Nuclear Safety Review System." The report further specifies that a log book of all changes to the criticality control portions of the configuration program is to be maintained for documentation and audit purposes. To prevent unauthorized changes, a physical audit of the criticality control section of the configuration program is to be performed once a year during the normal plant shutdown period. During that audit, a complete record of current configurations is to be generated from the actual operating configurations. This record is to be compared against the prior listings (those from the last audit) that have been maintained by Nuclear Safety Engineering Organization (NSE). The team requested copies of these audits. The licensee could not provide the record of the NSE's annual audit result. There is only one record, dated July 22, 1985, which was a letter requesting that the required audit be postponed until December 1985. There are not other records of the software configuration audit as described in Technical Report 3.13.18. The licensee has confirmed that no software configuration audit has ever been performed.

The team also requested an example of typical criticality control system-related documentation in the computer room log book. The team found that that log book contained the computer system software changes both for the criticality control and the process control-related changes. Many process control-related changes, such as disabling control room audible alarms, do not have Facility Change Request (FCR) documentation.

Based on an interview with the control engineer, the team found that the configuration control for the control system was weak. Management oversight to assure criticality safety controls is also lacking. For example, the control engineer was instructed by management to disable audible alarms in the control room control console because the operators complained that audible alarms were a nuisance. The control engineer was also instructed to remove the low interface interlock to the solvent-extraction system A column to prevent unnecessary interruptions of the process system. These changes were not reviewed by the nuclear safety engineering department, a finding verified by team interviews with nuclear safety engineers.

## 8.2 Nuclear Criticality Control Instrument System

### 8.2.1 Nuclear Criticality Control Functions in the Solvent-Extraction System

The instrumentation used in the digital computer control system as part of the nuclear criticality controls is identified as Active Engineered Control instruments (or "Essential

Instruments"). The instruments related to the following control functions are considered Essential Instruments.

1. Density Control on Feed Adjustment Tanks

   The objective of density control on the feed adjustment tanks is to limit the concentration of uranium in the feed adjustment tanks to less than a specified high limit. If density reaches the high limit, or if low flow or instrument failure occurs, the computer control system closes feed valves to terminate flow to tanks and alarms at the control console. Normal recovery from a condition of high density is the addition of dilution water.

   Density is measured using two different methods: direct measurement with a densitometer and use of a pressure transmitter and a level probe. The ratio of the pressure head to the level is the calculated density. Density measurements at the feed adjustment tanks (V-246, V-248) use Sarasota density meters. Density measurement at nitrate waste settling tank V-104 uses the calculated density method.

2. High Level Control at the Solvent-Extraction Carbonate Surge Tank (T-330)

   The objective of high level control is to prevent backflow to the ammonium carbonate make-up tank. When the tank reaches 85 percent of capacity, the computer control system closes inlet valves, alarms at the control console, and automatically transfers liquid out of tank T-330 to tank T-334.

3. pH Control at the Solvent-Extraction Carbonate Surge Tank (T-330) Solution

   The objective of pH control is to protect against uranium precipitation at either high or low pH levels. On detection of a high or low pH, the computer control system alarms at the control console and closes the carbonate feed valve, shuts off the pump, and opens the bypass valve to bypass solvent treatment until the pH problem is resolved.

4. High Level Control at the Solvent-Extraction Spent Solvent Surge Tank (T-334)

   The objective of high level control at T-334 is to control the amount of uranium going through the spent solvent stream. On detection of the 80-percent-full level, the computer control system alarms at the control console and bypasses the solvent treatment by opening the bypass control valve. If the level reaches the 90-percent mark, the drain valve is opened.

5. High Uranium in AQ Surge Tanks

   The objective of high ppm uranium control at the AQ surge tanks is to prevent the transfer from aqueous waste (AW) with high concentrations of uranium to the unfavorable geometry waste accumulation tank (V-103). When the laboratory sample results show concentrations greater than 150 ppm uranium, a control system interlock prevents the Q-tank dump valve (FV-420C) from opening.

### 8.2.2 Evaluation of the Nuclear Criticality Control Instrument System

Through interviews and plants tours, the team found that nuclear criticality control instruments were not clearly identified and that no apparent separation was made between instruments used for nuclear criticality control and those used for process control. No special signs or color codes existed to indicate the items for components and systems that were of nuclear safety significance. There are no redundancy designs, or special power source arrangements for these instruments; therefore, these instruments are not single-failure proof. The process control instruments and the nuclear criticality control instruments are in the same digital control system. Thus, any single failure could disable both process control and nuclear criticality control functions.

From interviews and document reviews, the team found that the Sarasota electronic converter experienced chemical corrosion problems in the field. About a year ago, the licensee installed protective boxes around the electronic components that solved the corrosion problem. The team reviewed maintenance work orders for two Sarasota density meters (DT-246, DT-248) which indicated that there were 16 repairs performed for DT-246 and 18 repairs for DT-248 during last three years. The licensee is, however, generally satisfied with the other instruments, including pressure transmitters, pressure switches, level switches, and level transmitters. A team review of maintenance work history for these instruments verified that repetitive problems did not exist. The evaluation of the Inductively Coupled Argon Plasma (ICAP) instrument sampling system is addressed in Section 10 of this report.

## 8.3 Instrument Calibration and Maintenance

### 8.3.1 Instrument Calibration and Maintenance Program

GE procedure 70-23 "Calibration Program for Essential and Non-Essential Instrumentation and Control," specifies the requirements for establishing an instrument calibration and maintenance program. The instrumentation used in the URU as part of the nuclear criticality control has been documented in GE technical report 3.13.17, "Critical Instrumentation Calibration." All essential instrumentation and controls are required to be calibrated every six months. The instruments for process control and monitoring functions are considered non-essential instrumentation and controls. Typically they are calibrated every 12 months, according to the process engineer. This was verified by a review of the calibration schedule.

Through interviews and document reviews, the team found that calibration is the only preventive maintenance performed for the instrumentation and control system. No periodic channel check or channel functional tests were performed.

The licensee's procedure for the instrument calibration program is as follows:

1. Specify instrumentation and control requirements.

2. Design, specify, procure, install, and test the system.

3. Operate the system.

4. Specify special calibration procedures, tolerances, and frequencies.

5. Perform calibration and maintenance.

Upon receipt of equipment and prior to installation, equipment requiring calibration is calibrated and tagged. After equipment is installed, the planner schedules the calibration according to the specifications. The planner prepares a work order for routine calibration. If the calibration requires that the process system shut down, the planner reschedules the calibration to minimize the period that the process is shut down. A routine calibration may be rescheduled from 15 days before to 15 days after the calibration due day. If the calibration still cannot be performed within that period, a "Measurement/Test Equipment QIE Non-Conformance" report is generated. (This report is signed by the maintenance department manager.) For unusual circumstances, routine calibration may be extended for up to 15 additional days. After that, the process system has to be shut down for the calibration. The non-conformance report is also used for other purposes, such as reporting instruments drifting out-of-tolerance limits, or when instruments cannot serve their intended function. A computerized Maintenance Improvement Program data base (MIPVAX) provides information for instrument calibration as well as data taken for other purposes. The MIPVAX generates an instrument calibration work order which provides calibration instructions and the forms the technician uses to record the calibration results. Included on the form is a reference range for that specific instrument. The technician has to fill in the "as-found" and "as-left" data when the calibration is completed. The calibration results are kept in the MIPVAX data base for future reference.

### 8.3.2 Evaluation of the Instrument Calibration Process

Through interviews and document reviews, the team found that the instrument calibration process has some weakness:

1. Although the calibration form generated by the MIPVAX system provides a reference range, it does not specify the calibration tolerance range for the instrument to be calibrated.

The technician does not have clear guidance on drift tolerance range, and does not know under what conditions instrument problems should be reported.

The team reviewed work order records for density meters DT-246 and DT-248. The following types of failures were typical: "wiring mistake," "component was valved out," "flow switch out of adjustment." However, no non-conformance reports were written for these instruments, nor was a review effort documented.

2. Although the planner reviewed calibration result sheets for the amount of time technicians spent on the work and whether work was completed on schedule, he showed no interest or knowledge in reviewing the "As Found" data. Therefore, instrument drift problems could go undiscovered.

3. The engineering units (e.g., milliamps and pounds per square inch) are not properly identified on the calibration data sheet. Although this format problem was known by the licensee since 1989, no effort has been made to correct it. The instrument technician supervisor told the team that technicians understood what the numbers represent because of their on-the-job training.

4. Although instrument calibration was performed by a team of two technicians, the result sheet does not require that a second person verify the calibration result. Nor is there an internal audit process to verify the accuracy of the data.

5. There is no engineering support to review or analyze the trends for the instrument failure data.

6. Technicians perform only routine calibration; they do not perform other preventive maintenance on instruments.

The team observed a typical instrument calibration on June 12, 1991. During that process, the technician had difficulty verifying a high alarm limit. Further investigation revealed that a wrong circuit board had been installed in the annunciator cabinet. A contact circuit board was mistakenly installed instead of an analog circuit board. A review of the previous calibration record was inconclusive in determining when and why the wrong circuit board was installed. This incident was a further example of the lack of work controls within the licensee's organization. The operators and the maintenance personnel are not routinely retrained to emphasize the effect their actions could have on nuclear safety.

## 8.4 Findings and Conclusions

- The digital computer control system is user friendly and reasonably reliable. There is no evidence that the digital computer control system hardware or software malfunctioned during this incident.

- The digital computer control system lacks formal software verification and validation documentation. There is no quality assurance program for software changes.

- There is no control or restriction relative to use of the TUNE mode operation at the control console. An unlimited manual override capability defeats the automatic control system safeguards.

- There is no sufficient engineering support to maintain the digital control system historical trend computer in reliable working condition. The process computer logging records were not retained for historical purposes.

- Configuration control implementation does not follow the licensee's own procedure. No software configuration audit was performed by NSE. Interlocks and alarms were removed from the automatic system when they became a nuisance to operations without a serious consideration being made of the criticality control implications of their removal.

- The criticality controls and the process controls are performed by the same digital control system. There is no separation between the criticality safety-related instruments and the process control instruments.

- There is not sufficient engineering support for instrument maintenance evaluations and instrument reliability analyses.

- The calibration activities are not documented thoroughly.

# 9 MAINTENANCE EVALUATION

This section evaluates the development and implementation of the maintenance program at the General Electric (GE) Fuel Manufacturing Operation (FMO) facility. The Incident Investigation Team evaluated three major areas of the program: (1) overall FMO facility performance related to maintenance; (2) management support of maintenance; and (3) maintenance implementation. In assessing facility maintenance, the team focused on electrical, mechanical, instrumentation and control, and the support areas of engineering, training, procurement, and operations. Particular attention was given to the connection each of these areas had to the events on May 28-29, 1991. This section also contains a detailed discussion of the licensee's post-incident disassembly, testing, and evaluation of level control valve LCV-300.

## 9.1 Overall FMO Facility Performance Related To Maintenance

Overall plant performance with respect to plant operability, equipment availability, and general plant reliability can be related directly to the effective implementation of maintenance activities. In assessing these activities, the team reviewed plant operations historic data, including the significant event precursors listed in Section 11 of this report, and monthly maintenance reports.

Three of the 15 significant precursor incidents listed in Section 11 were directly attributed to maintenance issues in general and are described below. These incidents revealed a weakness in the work control process. (Many of the other incidents listed in Section 11 indicated that maintenance activities may have contributed to the cause; however, their contribution could not be ascertained by the team because of the inadequacies in the licensee's root cause evaluations.)

- Nitrate Waste System - 1/12/90 - An improper gasket was installed in a valve in the nitrate waste system. As a result, a spill of approximately 50 to 100 gallons of nitrate waste occurred outside the Uranium Recycle Unit (URU) facility buildings. (See Section 11.3.7.)

- Radwaste System - 5/17/90 - A valve in a flow transmitter line was left closed after calibration. As a result, inaccurate quantities of liquid were collected in the URU radwaste daily composite sample. (See Section 11.3.4.)

- Fluoride Waste System - 2/5/91 - A manual valve in a flow control instrumentation loop was not reopened after calibration. As a result, approximately 33 kilograms (kgs) of uranium were transferred to an unfavorable geometry tank. (See Section 11.3.1.)

The monthly maintenance reports focused on the number of maintenance work orders (WO) accomplished according to WO categorization. Based on the licensee's reports, the maintenance program was balanced between corrective maintenance (CM) and preventive maintenance (PM), with the PM to CM ratio approximately 50/50. These reports did not address system or equipment availability related to maintenance. The licensee stated that obtaining and evaluating meaningful availability indicators would be a function of the Equipment Reliability Engineering group in the future. Equipment Reliability Engineering is discussed in Section 9.2.3 of this report.

## 9.2 Management Support of Maintenance

### 9.2.1 Management Commitment and Involvement

The degree to which plant management was committed to and involved in the maintenance process determined the degree of effort assigned in establishing, implementing, and improving maintenance. The maintenance organization at the GE FMO facility supported the operation of the plant; therefore, all priorities and resources appeared to focus on meeting the goals dictated by plant operations. Because of this philosophy, maintenance tended to be a reactive rather than a proactive organization. Based on WOs performed in the FMO from January 1991 through April 1991, 25 percent were categorized as Emergency/Breakdown WOs. This finding is discussed further in Section 9.3.1.

Management's recognition that maintenance control could be improved was evident with the initiation of a computerized Maintenance Improvement Program (MIPVAX) system in 1988 for planning and scheduling CM and PM maintenance activities. Management's ultimate goal was to integrate other computer systems with MIPVAX to obtain a virtually "paperless" WO system. The MIPVAX system was intended to retrieve maintenance work histories and integrate the spare parts inventory and procurement systems. Access on the inquiry mode was unrestricted. However, the MIPVAX program had two significant weaknesses: (1) no administrative governing procedures described the use and personnel responsibilities of the system and (2) maintenance work histories were not reviewed to detect repetitive failures and adverse trends. Although MIPVAX was potentially a complete and thorough system, the licensee admitted that it was relatively new and not utilized to its fullest capacity.

Management's philosophy towards PM was very narrow in scope. The team found that the majority of PMs performed at the plant were for instrument calibrations. Other PMs were limited to major pieces of equipment, such as pumps. In general, very little PM was performed on valves. Although many valves were identified in the plant and on piping and instrumentation drawings as "critical," no PMs or special maintenance instructions were identified. In many cases, a malfunction of these critical valves would allow transfer of uranium-bearing materials with concentrations higher than nuclear safety criticality limits from favorable to unfavorable geometry tanks.

### 9.2.2 Management Organization and Administration

Over one year ago, the FMO production maintenance department reorganized and eliminated direct supervision on all the shifts except for days (7:00 a.m. to 3:00 p.m., excluding weekends). The maintenance department operated on a four-shift (A, B, C, D) rotation, plus a steady day shift. The maintenance crews rotated from three 8-hour shifts with one crew off. There was one salaried supervisor for each of the six maintenance groups: (1) mechanical and electrical, (2) instrumentation, (3) control and instrumentation services (C&IS), (4) chemical, (5) heating, ventilation and air conditioning (HVAC), and (6) maintenance planners. Each shift (excluding days) had a total of eight maintenance workers from the first four groups. Each A through D shift had a maintenance coordinator who allocated labor according to priorities and assisted in controlling planned and scheduled work. However, the maintenance coordinators were neither salaried nor shift supervisors.

Because of the lack of management oversight on non-day shifts (including weekends), maintenance supervisors indicated that no sense of "ownership" existed on activities during these shifts. Little time was spent during turnovers between shifts discussing stages of maintenance accomplishment or troubleshooting activities. To compensate, many supervisors arrived hours early and stayed after their scheduled day shift to facilitate maintenance work transitions.

The groups that management had established for maintenance appeared to support the plant's maintenance requirements adequately. Although the plant had numerous specialized maintenance groups, the team noted "gray" responsibility areas within maintenance. Interviews with maintenance personnel indicated that maintenance on certain systems and components did not have clear cut responsibilities. Because of the gray areas, the potential for maintenance oversight existed. For example, the ordering of the replacement valve for LCV-300 was mechanical maintenance's responsibility; however, instrumentation maintenance had responsibility for the valve actuator. Instrumentation maintenance was never notified of the actuator replacement and, therefore, was not aware that new vendor manuals and calibration instructions were needed.

### 9.2.3 Technical Support

The team assessed the activities of the engineering organizations relative to maintenance support, modification installation, and material qualification. The team also evaluated the extent to which engineering principles and evaluations were integrated into the maintenance process.

Engineering organizations were not directly involved in the maintenance WO process. The process engineer could initiate a work request, which was developed into a WO by the maintenance planner; however, tracking the completion of a WO was up to the initiative of the engineer. Problems encountered during maintenance activities were usually handled by the maintenance department. If the problem was recognized to be beyond the knowledge of the maintenance staff, reliability/component engineers were called to resolve the problem. The process engineers were usually called if the problem still could not be solved by component engineers. Team interviews with maintenance personnel indicated that occasionally an engineering department representative would be called; however, problems were usually solved by maintenance rework.

Engineers were not required to review completed WOs on the computer system and, as a result, were not always aware of maintenance activities in the assigned systems or of trends involving equipment failures. To monitor adverse maintenance trends, perform root cause analysis, and analyze common mode failures, management established an Equipment Reliability Engineering Department approximately 6 months ago. Some of the department's functional responsibilities were: (1) define critical equipment, (2) determine its status, track its availability, and determine the root causes of its unavailability, (3) assist in establishing and evaluating spare parts and inventory requirements, (4) provide technical support, and (5) define calibration, preventive, and predictive maintenance for critical, new, and rebuilt equipment. This new program was initiated in the Fuel Component Operation (FCO) facility first, with the FMO facility to follow. Interviews with engineers in this department indicated that because of other priorities and the infancy of the program, the department was unable to devote the time necessary to perform all maintenance trending. As a result, adverse maintenance trends, such as the repetitive failures with LCV-300, were not noted, evaluated, and corrected.

The team reviewed the maintenance work history for LCV-300 (and the instrument loop) since its installation on December 26, 1990, as listed below. Four precursors to the valve malfunction on May 28, 1991, were identified and are marked (P) in the listing that follows. All precursor WOs were classified as Emergency/Breakdown. The problem and corrective action taken by maintenance are also described.

- 12/26/90 - WO 90-68567-00: Replace LCV-300 control valve on solvent-extraction (SX) Column A. No details were listed for the work performed other than that the new valve was located outside the mechanical maintenance supervisor's door.

- 12/30/90 - WO 90-69325-00: Install new conduit and calibrate. Instrumentation technicians indicated on the WO that this was a new valve and actuator installation and that no information was found for the new component. The WO did not contain any objective evidence that the calibration of the level control instrument loop was performed after installation of the new valve.

- 12/31/90 (P)- WO 90-69509-00: LIC-300 not controlling level. Maintenance set zero on the metering pump to correct the problem.

- 1/15/91 - WO 91-03673-00: Level transmission appears dead. Maintenance found solvent in the probe housing that ate the insulation off the wires.

- 2/13/91 - WO 91-09093-00: Repair damaged conduit and exposed wiring to level element LE-300, SX Column A level transmitter. Maintenance repaired both problems.

- 2/14/91 - WO 90-69460-00: Calibration of level indicating transmitter LIT-300, A Column Interface. This WO was originally scheduled to be performed on 12/31/90; however, it was rescheduled four times before being performed on 2/14/91. The reason given for the rescheduling was that the equipment was made unavailable by operations because the process was running and they did not want to shut down to perform the calibration. The loop calibration was completed on 2/14/91 with as-found readings within specifications. The rescheduling of calibrations is discussed further in Section 9.3.1.

- 3/24/91 (P) - WO 91-17000-00: LIC-300 is leaking through and will not maintain level properly. Maintenance documented that the positioner would not let the valve close because the open side would not bleed off when closing. An air line nut on the open side of the positioner was cracked opened to allow operation of the valve.

- 4/9/91 (P) - WO 91-20788-00: LIC-300 is not controlling as it should. The valve is staying open when it should be closed. The WO documented only 0.1 hour of maintenance, with no documentation of what, if anything, was accomplished.

- 4/10/91 (P) - WO 91-20862-00: Level-indicating transmitter LIT-300 indication to the process control system and local indication at the interface sightglass do not correspond. Maintenance could not find a problem; therefore, no corrective action was taken.

Other selected components in the SX and URU lab sample system were evaluated for adverse maintenance trends. The results are as follow:

- Since 1988, approximately 34 failures have occurred with two Sarasota density meters installed in the SX feed system. These density meters are an integral part of the uranium mass control system and isolate the solvent feed to the SX columns when high uranium concentrations are detected. These failures are discussed further in Section 8.3.2.

- Since 1988, approximately 33 percent of the WOs written for the URU lab sample system required the repair of multiport valves. No PM was initiated to preclude failures. Inadequacies in the lab sample system are discussed further in Section 10.

From this work history, it appears that maintenance's corrective actions were limited in scope, with no perspective to common mode failures or root cause analysis. Repairs were very specific in nature and did not look at failures from an overall plant perspective.

The engineering organization's involvement was evident in major system modifications through the Facility Change Request (FCR) process controlled by Practices and Procedures (P/P) 40-5, "Nuclear Safety Review System." An FCR was initiated for new or modified processes, equipment, or facilities, and assessed the need for a nuclear safety review. Other requirements, such as operating procedure changes, training for operations or maintenance, updating drawings, installation verification, spare parts list, and functional tests or calibration, were also listed on the FCR form for applicability. Under the current process, the decision as to whether or not a system or component modification rose to the level of requiring an FCR was left up to the person who initiated it, without an independent verification or review. Where work was limited to replacement of a component of "like kind," no FCR was required. Because of inadequate management oversight in implementing the procedure for FCRs, changes to equipment in the plant could be made without the process engineer's or management's knowledge until the equipment was installed in the plant. Besides bypassing potentially essential criticality safety evaluations, the failure to initiate FCRs resulted in less-than-adequate configuration controls. Instances where FCRs were not initiated, as required by the licensee's procedure, are listed below:

- In December 1990, LCV-300 was replaced with a similar actuating valve. The mechanical/electrical maintenance planner approved this change because the flow characteristics of the valve had not changed and the valve actuator, according to the vendor, was "similar" in function to the previous one. No FCR was initiated since replacement was considered, by the planner, to be of "like kind." Although similar in function, the new valve actuator was very different, in that actuation was an air-to-open, air-to-close function (spring-to-close with loss of air) using a positioner, as opposed to the previous air-to-open, spring-to-close function with no positioner. As a result, vendor manuals that specified the calibration procedure and spare parts were not obtained for the positioner and actuator. Failure to have the necessary manuals or spare parts for valve repair or replacement exacerbated the incident on May 29, 1991.

- An unauthorized installation of an upgraded fluoride liquid waste monitor was performed in December 1989. This installation complicated an incident involving the Fluoride Waste system as described in Section 11. Maintenance submitted a late FCR in January 1990; however, they incorrectly identified the monitor as a nitrate waste monitor.

- A fluoride analyzer in a refrigerated enclosure was installed under the V-108 tank at the Waste Treatment Facility without approval from Nuclear Safety Engineering.

This installation complicated an incident involving the Fluoride Waste system as described in Section 11.

## 9.3 Maintenance Implementation

### 9.3.1 Work Control

The team evaluated the effectiveness of the maintenance work control process to assure that plant safety, operability, and reliability were maintained. The team evaluated control of maintenance WOs, equipment maintenance records, job planning, the prioritization and scheduling of work, control of maintenance backlogs, maintenance procedures, post-maintenance testing, and completed documentation.

The maintenance WO control system was considered ineffective by the team largely because of weak administrative and management controls. Since the WO control process was not governed by any procedures, the system "control" consisted of a computer-generated WO form initiated from the MIPVAX automated processing system. Anyone was authorized to initiate a work request. The work request was then sent to the applicable maintenance planners and converted to a work order. Planning the work required that parts, work instructions, special work permits, and other supporting documentation be assembled before the work could be performed.

There were no required reviews or approvals to verify the WO's accuracy, resource availability, job priority, criticality safety requirements, or work scope. After WOs were put on the schedule, the maintenance shops were responsible for their completion in a timely manner. The authorization to begin maintenance was communicated verbally to operations. The status of equipment during or after maintenance work was performed was also communicated verbally.

Circumstances involving this and previous incidents, indicated that informal authorizations and statuses were inadequate. For example, because of an incident where a manual valve in a flow control instrumentation loop was not returned to an operating configuration after calibration (Section 11.3.1), steps had been added to calibration instructions for instrumentation technicians to notify the control room at the start and completion of calibrations. These steps were not apparent in the other maintenance department WOs.

The successful completion of maintenance tasks relied heavily on personnel experience. Detailed written guidelines for less experienced workers did not exist. The experience of the planner and the maintenance worker, as well as the use of vendor manuals, contributed the most to the completion of the WO. Although established maintenance procedures did not exist, the work directions for some instrument calibrations and repetitive WOs appeared consistent and were somewhat more detailed than Routine or Emergency/Breakdown WOs.

After maintenance was completed, no formal post-maintenance tests were performed. Team interviews with maintenance workers indicated that usually a component functional check was performed to confirm that the problem was corrected before the component was returned to operations; however, operations did not seem to require the performance of a system operability test to verify that the system functioned properly after maintenance was completed. When the WO was completed, the worker added any appropriate comments and closed out the WO on the computer. Except for calibrations, the hard copy of the WO was discarded. The licensee stated that a hard copy of the WO may be provided to the worker's supervisor but was not required. The monthly maintenance reports were the only indication that the WO computer data was being used for tracking or trending.

A team review of completed WOs indicated that workers were inconsistent in their use of the comment section. Most of the WOs had little, if any, comments on as-found conditions of equipment and descriptions of work performed. However, comments were noted when operations would not release the equipment for maintenance work. Also, once the worker completed the WO, no one was required to review it. Because no reviews were performed after work was completed, no formal root cause determination was performed. The consequences of unreviewed WOs were discussed in Section 9.2.3.

Provisions were in place to perform emergency maintenance using an Emergency/Breakdown classified WO. Emergency/Breakdown WOs bypassed the maintenance planner and were written directly on the computer using the MIPVAX system and sent to the applicable maintenance shop. Although it is expected that emergency repairs are needed, the team determined that this method sometimes appeared to be inappropriately employed by the control room operators when non-emergency equipment malfunctions affected system performance or threatened system shutdown. These Emergency/Breakdown WOs did not contain work instructions but merely described the problem. The workers had the responsibility to see that the problem was corrected using troubleshooting techniques or vendor manuals. Since no maintenance supervision was on site except during day shifts, the priority of work appeared to be dictated by the operations department with Emergency/Breakdown WOs. When priority conflicts arose, the area coördinator determined the priority. On non-day shifts, planned WOs appear to be "filler" work performed in between Emergency/Breakdown WOs.

The Production Maintenance organizations (FMO, FCO, and SCO) had a backlog of 2374 WOs as of June 10, 1991. According to the licensee, this backlog included both corrective and preventive (inspection and calibration) maintenance WOs and represented approximately six weeks of work utilizing the current maintenance staff. The backlog for the FMO was divided into the following categories:

| BACKLOG | CREW NAME | ESTIMATED TIME TO COMPLETE* |
|---|---|---|
| 185 | FMO HVAC | 3-4 weeks |
| 450 | FMO C&IS | 4 weeks |
| 216 | FMO Instrumentation | 3 weeks |
| 292 | FMO Production Maintenance | 1-2 weeks |
| 673 | FMO Production Support | 5-6 weeks |

*Assuming all material on hand and no new WOs issued.

The backlog did not appear to be excessive and was within the capabilities of the current maintenance staff. Approximately 54 percent of the backlogged WOs were classified as PM. However, based on work orders performed in the FMO from January 1991 through April 1991, 25 percent were categorized as Emergency/Breakdown WOs, with only 11 percent categorized as routine planned maintenance. The team determined that many WOs were inappropriately characterized, as evidenced by the large percentage and age of Emergency/Breakdown WOs in the backlog. WOs involving preventive maintenance inspections represented 51 percent of the completed maintenance work for that same period.

The team reviewed completed PMs and calibrations performed on the aqueous waste (AW) and SX systems since 1989 and determined that 22 percent of the SX PMs and 17 percent of the AW PMs had been rescheduled three times or more. Rescheduling PMs four to five times was not uncommon, with one PM rescheduled 18 times. Although this appeared excessive, very few essential calibrations exceeded the past due date. Other equipment PMs were not considered essential and did not have an absolute due date. If a monthly PM was not performed one month, the next's month's PM would replace it; however, this practice was not excessive. According to maintenance management and planners, the majority were rescheduled because operations would not allow the systems to be shut down or put in a configuration that would accommodate the maintenance task. The team concluded that the large number of rescheduled PM items resulted from maintenance scheduling inadequacies combined with the operations philosophy of "keep the process running."

### 9.3.2 Maintenance Accomplishment

The team reviewed numerous completed WOs to determine if the maintenance specified was accomplished as required. This review included an evaluation of qualified personnel, proper prioritization of work, quality of documentation for component work histories, description of problems and their resolutions, and post-maintenance testing. A detailed evaluation focused on two WOs performed on LCV-300 on May 28, 1991, and May 29, 1991. Since documentation of the work performed was not detailed, the team gathered data through interviews with the instrumentation technicians assigned the WOs.

- WO 91-30879-00: This WO, performed 5/28/91 on the 11:00 p.m. - 7:00 a.m. shift, was written by the instrumentation technicians after the maintenance had been completed. A WO was never initiated, as requested by the technicians, from CRO A. Instrumentation technicians were called by CRO A and informed that LCV-300 had a problem that needed immediate attention. For reasons unknown, the technicians were unable to respond until two hours later. When they did respond, they found the valve stuck in the open position. After troubleshooting, the problem was determined to be in the Moore positioner. The positioner was disassembled, cleaned, and reassembled in the field. As noted in previous WOs, the instrumentation shop did not have a vendor manual for this positioner. When a demand signal was sent to LCV-300, the valve stuck closed. Again the positioner was disassembled. During both disassembles, the technicians indicated that parts from the positioner fell out onto the floor. (Based on the disassembly of the positioner on June 20, 1991, all parts appeared to be correctly in place; however, the span-adjusting screw and locknut were not tightened correctly. Details of the disassembly are discussed in Section 9.4.) Replacement parts were then sought; however, they could not be located using the MIPVAX system. The technicians informed CRO A that the valve could not be repaired. CRO A then requested that the valve be forced open to facilitate running the SX system manually by the operator on the floor. To accommodate his request, the technicians manipulated the existing air lines to the bottom of the Annin actuator to maintain the valve in the open position. Temporary maintenance such as this had been performed on this valve on March 24, 1991, under WO 91-17000-00, as discussed in Section 9.2.3. Interviews with the technicians indicated that temporary modifications beyond the normal function of the equipment were rarely performed. However, the SX process was extremely high priority because of complications involved in starting the system once it was shut down. Because of the lack of documentation on completed WOs, the team could not determine the frequency of temporary repairs.

- WO 91-30879-00: This WO documented maintenance performed on May 29, 1991, during the 7:00 a.m. - 3:00 p.m. shift after the SX system was shut down. Technicians found LCV-300 configured with a piece of poly flow tubing supplying constant air pressure to the actuator. This allowed the valve to remain in the open position. The Moore positioner was removed from the valve and taken to the instrumentation shop. When the positioner was disassembled, one technician noted a greasy residue similar to a moisture, air, and oil mixture on the spool piece. Once cleaned, the positioner was reassembled and bench tested. The positioner functioned properly, so the unit was reinstalled on the valve in the plant and again functionally tested. The LCV-300 valve assembly responded correctly to the signals produced from a current generator.

Further investigation by the IIT into LCV-300 revealed the following:

(1) The LCV-300 Stores File Inquiry prior to the valve replacement on December 26, 1990, indicated "no substitutions" under special instructions. Since the incident, the licensee added a note that instrument engineering's review and approval were needed when ordering a valve.

(2) The LCV-300 replacement valve was purchased without the procurement of spare parts or replacements for the positioner or actuator. The work history showed that WO 90-69325-00 on 12/30/90 indicated that the new valve assembly had not been reviewed by instrumentation maintenance. The lack of a vendor manual made installation and calibration difficult. WO 90-69325-00 documentation showed no objective evidence that the valve was calibrated after installation.

(3) This valve actuator assembly was described by maintenance personnel as a "one of a kind," "first time ever seen" in the plant; therefore, the detail required for the initial setup and calibration appeared very complex without a vendor manual. This difficulty was verified when the Masoneilan representative assisted in the setup and calibration of the valve with the new Moore positioner. This process is discussed further in Section 9.4.

Disassembly and testing of LCV-300 is discussed in Section 9.4.

### 9.3.3 Maintenance Materials Control

Obtaining parts for maintenance was accomplished with the MIPVAX computer system. There were established minimum and maximum stock limits on most items. Based on the WOs on backlog because of parts unavailability (less than 3 percent), obtaining spare parts did not appear to be a problem. Maintenance requested, by computer, certain parts that were needed for a WO. The warehouse charged the parts to a WO number and sent the parts where directed. However, spare parts were not on site for LCV-300 because none were ordered when the new valve was purchased.

### 9.3.4 Maintenance Personnel Training and Staffing Requirements

The team evaluated the maintenance training and staffing requirements to determine the extent to which both affected the maintenance process. Training requirements for maintenance personnel in the URU consisted of: (1) basic nuclear safety training, (2) emergency response team training (fire brigade) for all volunteers, (3) fire extinguisher training, (4) hazards communication training, (5) industrial training, and (6) anemometer and digital scale training for designated HVAC personnel. There was no formal training required specific to the maintenance areas. The majority of the training received was on-the-job skill attainment, with occasional vendor training provided on specific equipment. As discussed previously, the lack of formal training appeared to be compensated for by the extensive personnel experience and was relied upon heavily for maintenance performance. Since many of the maintenance staff will be eligible for retirement in 10 to 12 years, management would be faced with difficult succession planning using the current training requirements. Increased management attention in this area was evident with the following programs:

- The maintenance organization initiated a training program May 19, 1989, targeted at basic adult education in response to lower-than-expected results to a basic comprehension skills evaluation. Refresher training was conducted in fundamental areas on site with support from the local community college. Approximately 75 of 126 maintenance personnel have completed the training. Management anticipated that the successful completion of this training would result in an increased understanding and knowledge of maintenance manuals used to repair equipment.

- The site had recently initiated a certification program February 1990 for apprentice-type training, following the successful completion of the refresher training. The training consisted of approximately 11 classes that included blueprint reading, basic electricity 1 and 2, and basic instrumentation. The program was still in the initial stages of implementation.

The success of the entire maintenance organization relied more on the experienced, stable work force than on established maintenance procedures and programs. The stability, because of low personnel turnover, resulted in a maintenance staff that had from 10 to 23 years experience at the plant in various capacities within the maintenance organization. This experience compensated for the vulnerabilities noted in the work control process discussed in Section 9.3.1.

Employment practices throughout the plant were focused on not increasing the number of employees. Because of this, maintenance job openings were filled from within the existing licensee's plant staff. This philosophy tended to force on-the-job training as the principal means of attaining minimum qualifications, as opposed to hiring employees with maximum qualifications from outside the existing staff. According to the maintenance manager, the existing practice posed a particular problem in obtaining qualified electricians and instrumentation technicians.

## 9.4 Maintenance Post-Incident Analysis of LCV-300

Because the failure of LCV-300 was the initiating contributor to the incident, the U.S. Nuclear Regulatory Commission (NRC) Incident Investigation Team (IIT) requested that the licensee disassemble the valve and supporting components for testing and evaluation.

### 9.4.1 Component Description

The SX A Column interface level is maintained with an instrumentation loop. The components in the loop consist of a level-sensing element, a level transmitter, and a level control valve, LCV-300.

Level control valve LCV-300 consists of a Masoneilan 1-inch globe valve with a direct-acting Moore positioner and a double-acting Annin actuator (Figures 3.6 and 9.1.). (The valve and valve components were received from Piedmont Instruments, which distributes the LCV-300 valve assembly for the Masoneilan Valve Company.) The Moore pneumatic valve positioner directs air flow to the top or bottom of the Annin actuator piston, which is designed for on-off or throttling service.

### 9.4.2 Failure Mechanism Analysis for LCV-300

On June 8, 1991, the licensee developed the following scope of activities for testing and evaluation of LCV-300:

- Document as-found conditions and review all known information and data defining conditions prior to, during, and after the incident.
- If possible, test the valve under the conditions during which equipment may have failed to operate correctly.
- Identify any apparent root causes of equipment malfunction.
- Document all findings, retaining any failed/misplaced equipment for subsequent review.

#### 9.4.2.1 Troubleshooting Activities and Results

On June 20, 1991, the licensee assembled a team, which included a representative from the Masoneilan Valve Company, to determine the potential components whose failure would fit the malfunctions observed for LCV-300 on May 28-29, 1991. The team tested and evaluated the valve both in the system and, after removal from the system, in the instrumentation shop.

Prior to testing the valve in the system, the team visually inspected the valve to document its as-found condition. The only anomaly noted was a constant air bypass noise emitted from the Moore positioner. The following tests were performed with the valve in the system to verify the fail-safe (close) mode and complete movement of the valve.

1. Tests of air supply failure modes
    a. Air supply valve closed.
    b. Air supply removed from positioner.
    c. Air supply removed from the electropneumatic transducer (I/P).
    d. Air supply pressure verified.
    e. Valve closure with existing air supply verified.

2. Verification of valve closure upon removal of electronic (4-20 milliamps (ma)) input to I/P.

3. Verification of full closing and opening of valve and linearity of movement with a 4-20 ma input source to the I/P.

In all test steps, the valve closed as designed with no anomalies noted. No binding or other motion-inhibiting phenomena were observed during the stroking of the valve. As per design, linearity results indicated 8 ma = 1/4 open, 12 ma = 1/2 open, 16 ma = 3/4 open, and 20 ma = (approximately) full open. The same results were obtained in the closing direction.

The valve was removed from the process line and moved to an instrumentation shop for the remainder of the testing and disassembly. Once set up on the test bench, the same tests for valve failure modes were performed as in the system. Air supply failure modes were tested, with the valve closing as designed. However, when the 4-20 ma source input was applied to the I/P with the valve fully closed, the valve did not open as designed. The valve remained in a stuck-closed position that appeared similar, if not identical, to one of the malfunctions observed on May 28-29, 1991. To investigate the cause of the anomaly, the team disassembled the valve and actuator components.

### 9.4.2.2 Moore Positioner

The Moore positioner was disassembled with no foreign materials noted in the positioner housing (Figures 9.2 and 9.3); however, the span-adjusting screw and locknut were loose (Figure 9.4). The Masoneilan representative indicated that this condition would not likely result in the valve malfunctioning. Disassembly of the spool housing assembly revealed that the end cap diaphragm was adhering to the spool housing and the spool piece itself. Impressions of the spool piece could be distinctly seen in the neoprene diaphragm (Figure 9.5). An oily, wet film was also noted at the spool and diaphragm interface. After removal of both the pilot and end cap diaphragms, the spool piece could not be manipulated and appeared to be stuck. With considerable force applied by hand, the spool was freed and removed. Examination revealed that the upper collar on the spool appeared to have been manually filed down and was out of round. Small grooves (possibly from vise grips) were also observed on the tip of the spool shaft (Figure 9.6). With simulated operation, the spool easily became stuck inside the bushing in the spool housing assembly. The brass bushing was also slightly scored indicating that interference between the two had occurred at some point in time. According to the Masoneilan representative, if the spool was not free to move inside the spool housing assembly, the positioner would be unable to control the valve actuator by properly porting and exhausting air. The representative also indicated that the constant air bypass noise emitted from the Moore valve positioner out in the plant resulted from the filed down spool because spool and bushing tolerances were no longer correct and allowed air to pass between them. Review of maintenance work history for LCV-300 did not indicate that the spool had been filed; however, very little work performance documentation was found on any work order performed on LCV-300.

An examination of the spool assembly housing showed that two small weep holes had been partially blocked by a soft, unidentified substance (Figure 9.7). Also noted was the use of Teflon tape on the instrument line fittings (Figure 9.5). The Masoneilan representative indicated that the weep hole blockage could also prevent the positioner from performing properly. However, because the valve had operated correctly in the system tests, this blockage did not appear to be the root cause.

#### 9.4.2.3 Annin Actuator

The Annin Actuator was the next component removed and disassembled. No anomalies were noted during the disassembly of the actuator and the O-rings and the silicon grease appeared new (Figure 9.8).

#### 9.4.2.4 Masoneilan 1-inch Globe Valve

The Masoneilan globe valve was disassembled next. Inspection of the plug and stem assembly and the seat ring revealed small grooved markings as if the plug had, at one time, closed on an obstruction. The Masoneilan representative indicated that it was not unusual to find small markings of this type on a new valve. Particles from the manufacturing of the valve could have fallen off during valve use. The licensee machined the plug and stem assembly and the seat ring to obtain a good seating surface. Since the valve had only been installed in the system for approximately six months, all packing and gaskets appeared new.

#### 9.4.2.5 Air Pressure Regulators

The two air supply regulators and filters were also disassembled. Inspection showed that both the filters and the regulators were in good condition. Slight evidence of oil and foreign materials was found in the housings. However, during plant and system walkdowns, the team noted that the instrument air lines between the air dryer and most instrumentation were made of carbon steel. Because of the importance of clean, dry, oil-free instrument air, the team requested information regarding the particle size that could be filtered by the various filters in the URU instrumentation system and the LIC-300 (LCV-300) control loop. The following data was provided:

- Instrument air dryer: 1st stage prefilter = 0.3 micron
  2nd stage prefilter = 0.01 micron
  After filter = 0.3 micron
- Filter at LCV-300: 5 microns

The vendor manual for the Moore positioner states that the maximum particle size in the air stream at the instrument should be no larger than 3 microns. Failure to have clean air would increase the possibility of a malfunction or deviation from specified performance. The licensee stated that the filters recommended by the manufacturer were used at the instrument air dryer. However, the filter at the valve allowed larger-sized particles to pass to the instrument than was recommended by the vendor. Particles of this size were probable because of the presence of carbon steel piping between the air dryer and the valve.

#### 9.4.2.6 Root Cause Evaluation

The licensee and the Masoneilan representative, reassembled the valve with a new Moore positioner and the existing Annin actuator. The valve assembly was then calibrated with guidance from the Masoneilan representative.

The valve assembly was reinstalled in the system and again tested in the same manner as the initial as-found condition. The valve performed as designed.

On June 26, 1991, a representative from Moore Products evaluated the disassembled positioner to determine possible root causes for its malfunction. Although the exact root cause of this event could not be ascertained, the representative stated that based on the scored brass bushing, it appeared that prior interference between the spool and the bushing had occurred and an erroneous root cause evaluation resulted in filing the spool. The representative verified that the spool was not filed down at the Moore factory.

The licensee planned to continue its investigation of LCV-300 by: (1) reviewing maintenance and calibration histories, (2) assessing previous problems that may have relevant bearing, (3) reviewing data from vendors, operators, and maintenance, and (4) evaluating relevant operation prior, during, and after the incident on May 28, 1991. Based on the data collected, hypotheses would be developed to identify the most likely root cause of the valve malfunction. The licensee told the IIT that it planned to complete the investigation and summarize its findings in late July 1991.

### 9.4.3 General Observations

General observations noted by the IIT during the disassembly of the valve and components were as follows:

1. The most likely root cause appeared to be the stuck spool in the Moore positioner. The Masoneilan and Moore Products representatives stated that procurement of the Moore positioner with an out-of-round spool (manually filed down) was highly unlikely.

2. The filter at LCV-300 allowed particles (of 5 microns) larger than those recommended by the manufacturer (3 microns) to pass to the positioner. Particles of this size were probable because of the carbon steel piping between the air dryer and the valve. Failure to have clean air would increase the possibility of a malfunction of the positioner.

3. Teflon tape used during the assembly of the components tends to migrate into the instrument components and could cause blockage of the small air ports. The team was informed that the instrument maintenance shop used Teflon tape almost exclusively.

4. The detail required for the initial setup and calibration of the Moore positioner and valve assembly appeared very detailed and complex. Without the assistance of the Masoneilan representative, or the use of the vendor manual, correct setup and calibration appeared difficult. From the time when the valve assembly was installed on December 26, 1990, the instrumentation shop did not have a copy of the vendor manual for the Moore positioner.

5. Preventive maintenance, other than a yearly calibration, was not performed on LCV-300.

### 9.4.4 Findings and Conclusions

This section highlights the most significant findings and conclusions of the team's assessment of the licensee's maintenance program. The most significant weaknesses occurred in the area of management involvement, maintenance work control, and engineering support. Specific elements were:

<u>Management Involvement</u>

- Maintenance priorities and resources focused on meeting continued plant operation goals.

- No administrative governing procedures described the use and personnel responsibilities of the maintenance process.

- Maintenance supervision on shifts other than days was lacking.

<u>Maintenance Work Control</u>

- Explicit work instructions in work orders were lacking.

- Work documentation was inconsistent and did not address as-found or as-left information.

- Formal post-maintenance testing was not performed.

- There were scheduling inadequacies that resulted in numerous preventive maintenance items being rescheduled.

- There was a lack of formal authorization for work initiation, progress, and completion.

- Temporary maintenance repairs were performed to support continued operation.

- Work orders lacked reviews before or after work was completed.

Engineering Support

- There was a lack of engineering involvement in the maintenance process, especially in the Facility Change Request process.

- Maintenance trending to detect adverse maintenance conditions and analysis of root cause and common-mode failures were not performed.

These weaknesses, individually or combined, have resulted in the following general conclusions:

- Maintenance work control problems were directly attributable to 4 of 15 safety-significant plant incidents (including this incident).

- Because of inadequate management oversight in implementing the Facility Change Request procedure, some facility and component changes were made without management or nuclear safety reviews. Three such changes, including the replacement of LCV-300, complicated significant plant incidents.

- Because maintenance task completion relied heavily on the skill and experience of craft personnel and on scheduling affected by continued plant operation, improper repairs or parts were sometimes utilized. For example, the spool in the Moore positioner was erroneously filed down.

- Because completed maintenance work was not reviewed, adverse maintenance trends (such as the repetitive failures with LCV-300, the Sarasota density meters, and the URU laboratory multiport valves) were not identified, evaluated, or corrected.

In general, the success of the licensee's entire maintenance organization relied more on its experienced, stable work force than on established maintenance procedures and programs. Management's inadequate oversight of the maintenance process, coupled with the facility's production-oriented philosophy, resulted in affording maintenance personnel with extensive work latitude without specifying the impact that maintenance activities could have on criticality controls.

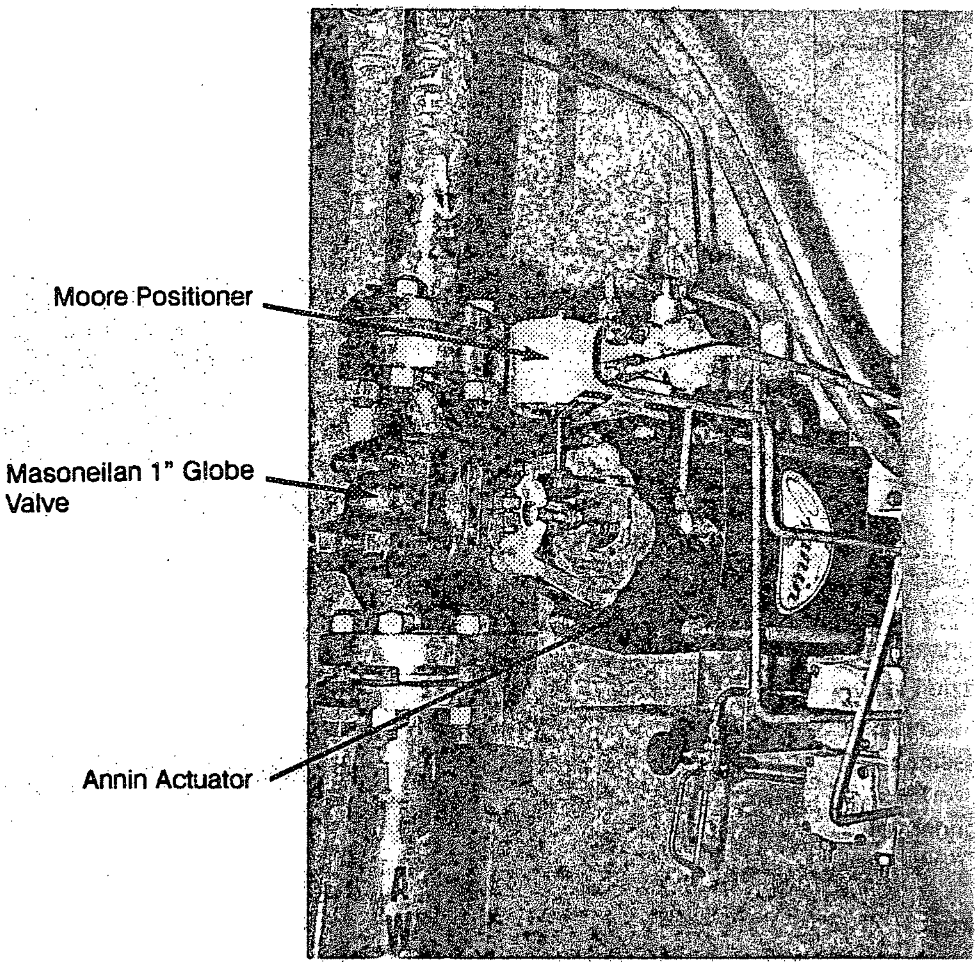



Figure 9.1 LCV - 300

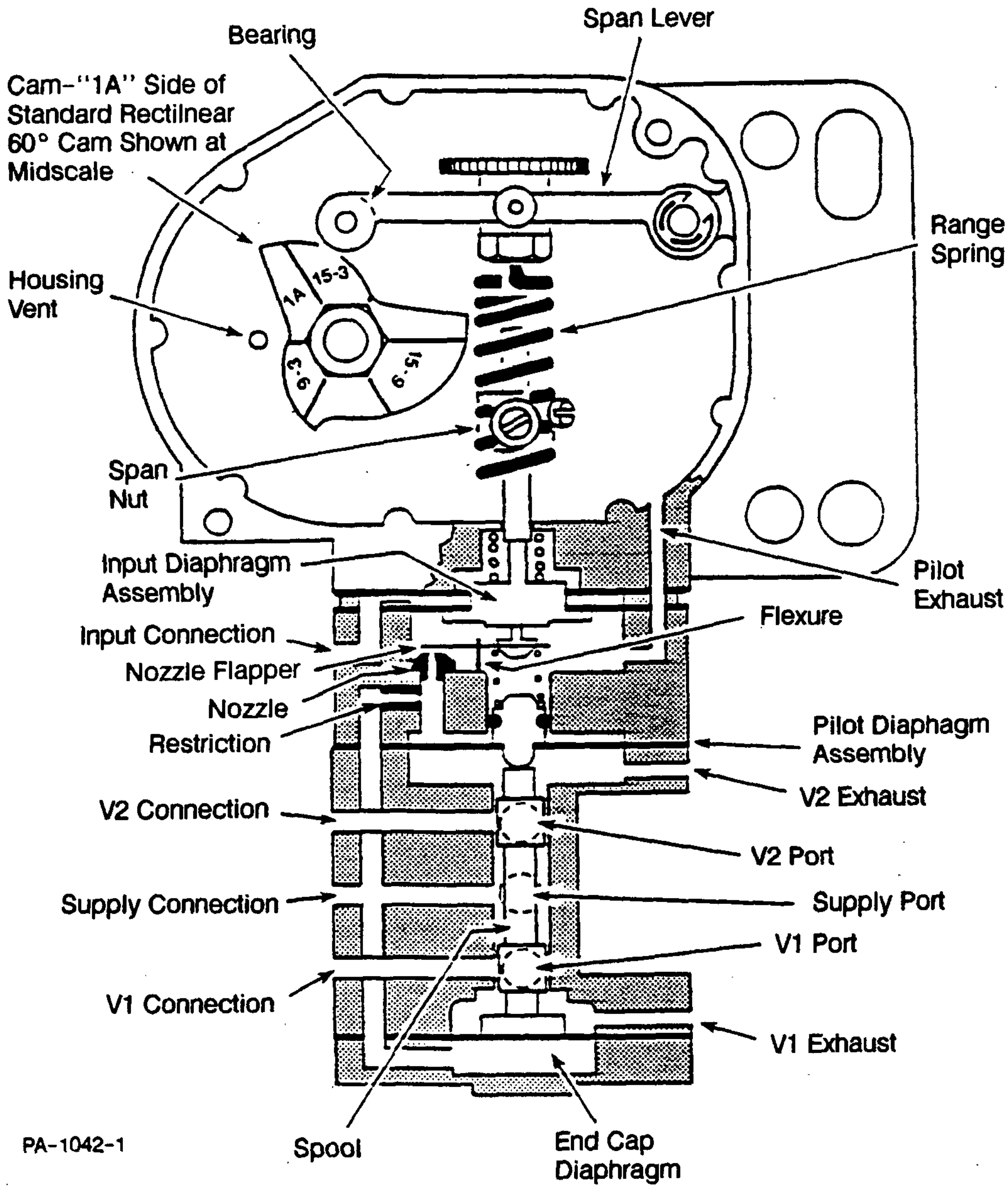
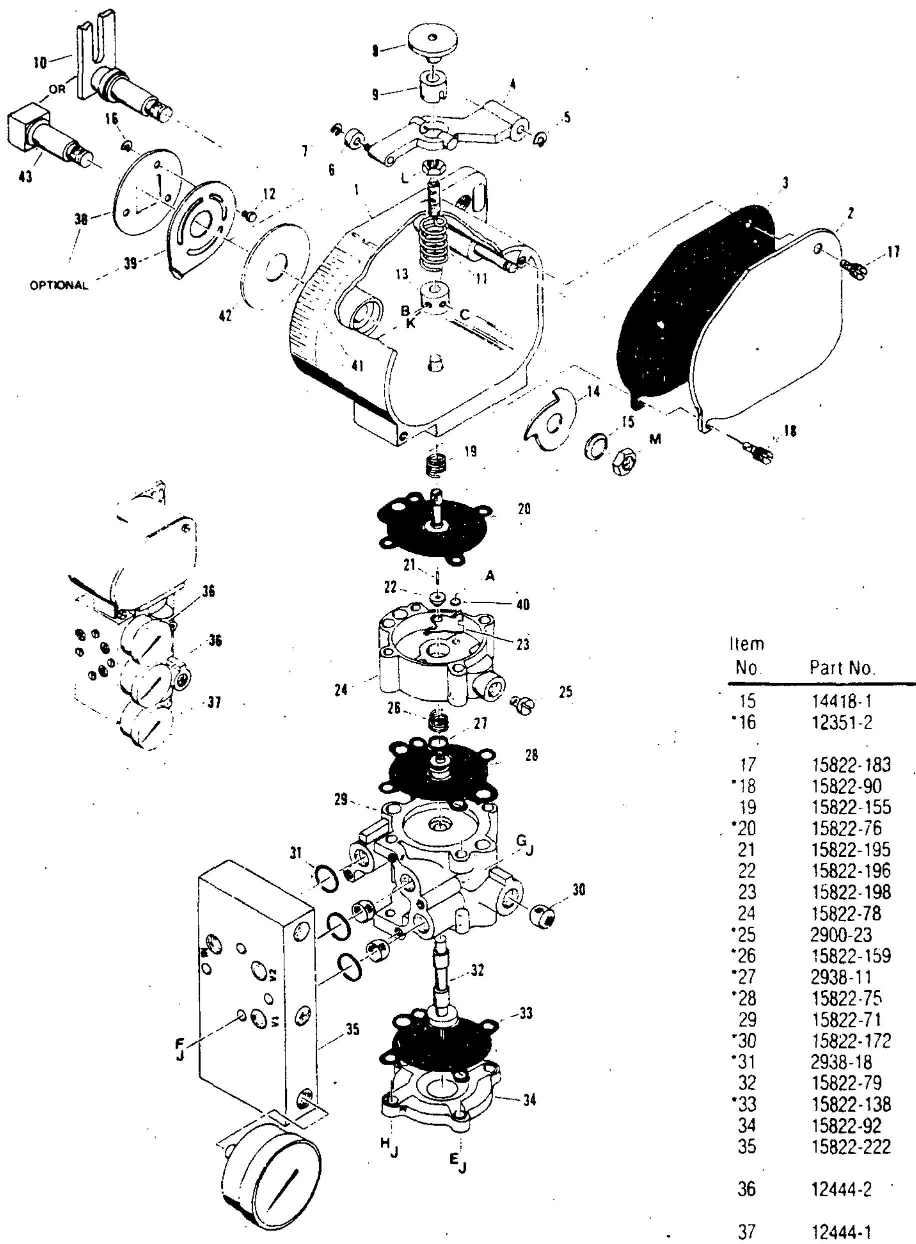



Figure 9.2 Moore positioner




| Item No. | Part No. | Description | Req'd. |
|---|---|---|---|
| 1 | 15822-70 | Housing Assembly | 1 |
| 2 | 15822-137 | Housing Cover | 1 |
| *3 | 15822-217 | Cover Gasket (Optional) When Specified | 1 |
| 4 | 15822-89 | Span Lever Assy. (Incl. Items 6 & 7) | 1 |
| *5 | 6773-5 | E-Ring | 1 |
| 6 | 15822-85 | Bearing | 1 |
| *7 | 6773-7 | E-Ring | 1 |
| 8 | 15822-147 | Thumbwheel | 1 |
| 9 | 15822-77 | Zero Screw Seat | 1 |
| 10a | 15822-209 | Pivot Arm, 6" (For Use With Rectilinear Actuators Only) When Specified | 1 |
| 10b | 15822-207 | Pivot Arm, 4" (For Use With Rectilinear Actuators Only) When Specified | 1 |
| 11 | 15822-149 | Range Spring | 1 |
| *12 | 15822-226 | Indicator Pin (Optional: Used With Items 16, 38, 39 & 43 Only) When Specified | 3 |
| 13 | 15822-74 | Zero Nut | 1 |
| 14a | 15822-152 | 90° Characterized Linear Cam (Used With Rotary Actuators Only) When Specified | 1 |
| 14b | 15822-162 | 60° Characterized Linear Cam (Used With Rectilinear Actuators Only) When Specified | 1 |
| 15 | 14418-1 | Washer | 1 |
| *16 | 12351-2 | E-Ring (Optional: Used With Items 12, 38 39 & 43 Only) When Specified | 3 |
| 17 | 15822-183 | Holding Screw | 1 |
| *18 | 15822-90 | Cleaning Wire (And Holding Screw) | 1 |
| 19 | 15822-155 | Suppression Spring | 1 |
| *20 | 15822-76 | Input Diaphragm | 1 |
| 21 | 15822-195 | Pivot Pin | 1 |
| 22 | 15822-196 | Spring Seat | 1 |
| 23 | 15822-198 | Nozzle Flapper | 1 |
| 24 | 15822-78 | Pilot Ring Assy. (Incl. Items 23, 25, 40 & A) | 1 |
| *25 | 2900-23 | Sealing Screw | 1 |
| *26 | 15822-159 | Gain Spring | 1 |
| *27 | 2938-11 | O-Ring | 1 |
| *28 | 15822-75 | Pilot Diaphragm Assy. (Incl. Item 27) | 1 |
| 29 | 15822-71 | Spool Housing Assy. (Incl. Item 30) | 1 |
| *30 | 15822-172 | Screen Insert | 3 |
| *31 | 2938-18 | O-Ring (Optional: Used With Item 35) | 3 |
| 32 | 15822-79 | Spool | 1 |
| *33 | 15822-138 | End Cap Diaphragm | 1 |
| 34 | 15822-92 | End Cap | 1 |
| 35 | 15822-222 | Manifold Block (Optional: Incl. Item 31) When Specified | 1 |
| 36 | 12444-2 | Pressure Gauge, 0 to 160 psig (Optional) When Specified | 2 |
| 37 | 12444-1 | Pressure Gauge, 0 to 30 psig (Optional) When Specified | 1 |
| 38 | 15822-225 | Plate (Optional: Used With Items 12, 16 39 & 43 Only) When Specified | 1 |
| 39 | 15822-224 | Pointer (Optional: Used With Items 12, 16 38 & 43 Only) When Specified | 1 |
| *40 | 12372-38 | Washer | 2 |
| 41 | 15822-237 | Indicator Scale (Optional: Used With Items 12, 16, 38, 39 & 43 Only) When Specified | 1 |
| 42 | 15822-181 | Vent Shield | 1 |
| 43 | 15822-42 | Input Shaft (For Use With Rotary Actuators Only) When Specified | 1 |
| | | Hardware | |
| *A | 1-0225 | #2-56 x 3/16 Soc. Hd. Screw | 2 |
| B | 1-0558 | #4-40 x 3/16 Fill. Hd. Screw | 1 |
| C | 1-0605 | #4-40 x 1/4 Fill. Hd. Screw | 1 |
| E | 1-1921 | #8-32 x 1/2 Fill. Hd. Screw | 2 |
| F | 1-2085 | #8-32 x 1-1/4 Fill. Hd. Screw | 4 |
| G | 1-2120 | #8-32 x 1-1/2 Fill. Hd. Screw | 2 |
| H | 1-2190 | #8-32 x 3-1/4 Fill. Hd. Screw | 2 |
| J | 1-7274 | #8 Lockwasher | 10 |
| K | 1-7425 | #4 Flatwasher | 1 |
| L | 1-7723 | #10-32 Hex Nut | 1 |
| M | 1-7740 | 1/4-20 Hex Nut | 1 |

Figure 9.3 Assembly of Moore positioner

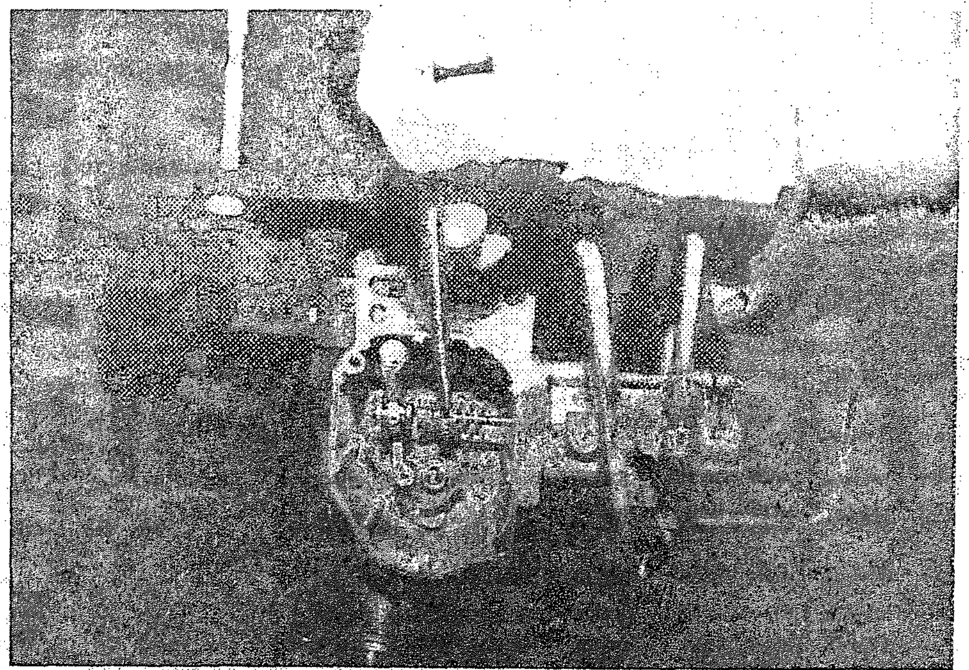


Figure 9.4 Moore positioner with pointer indicating loose span adjusting screw and locknut

Teflon tape

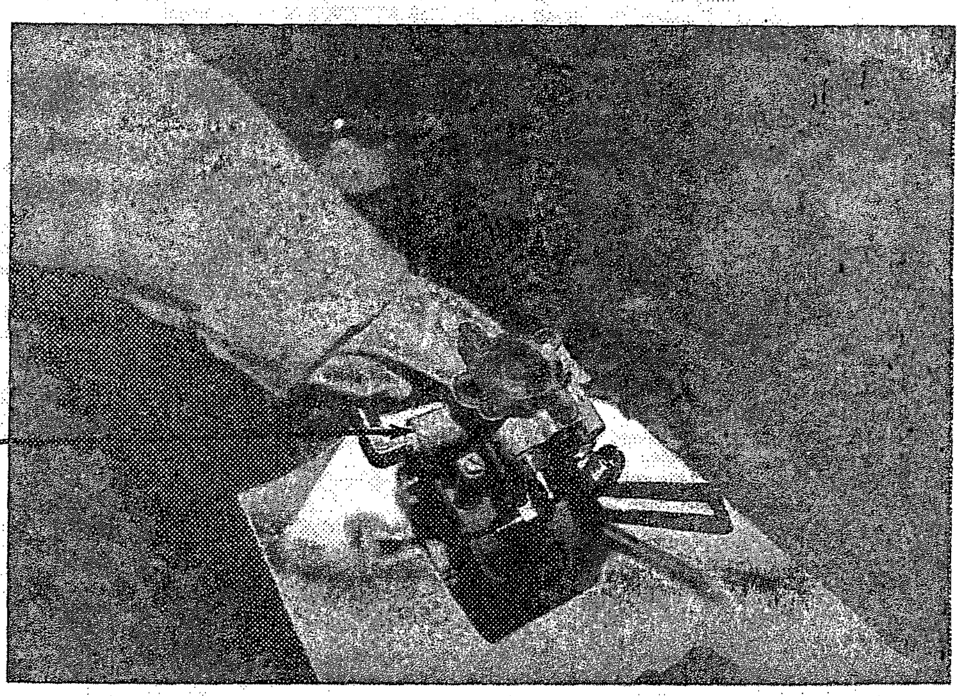


Figure 9.5 Moore positioner with end cap removed

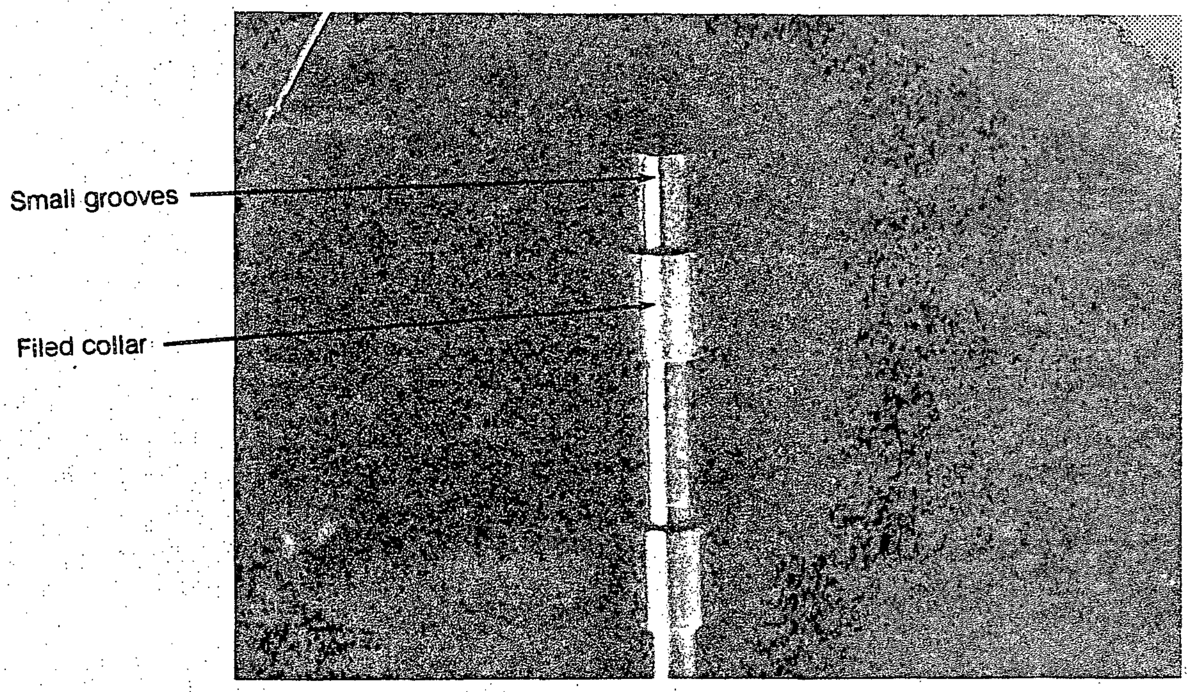



Figure 9.6 Moore positioner spool

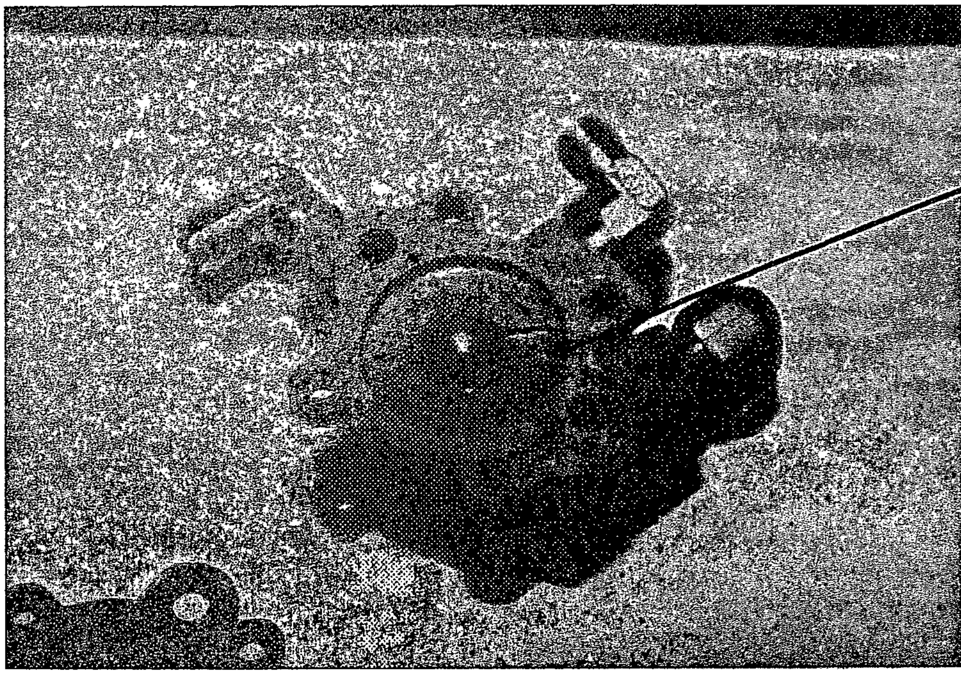



Figure 9.7 Spool housing assembly

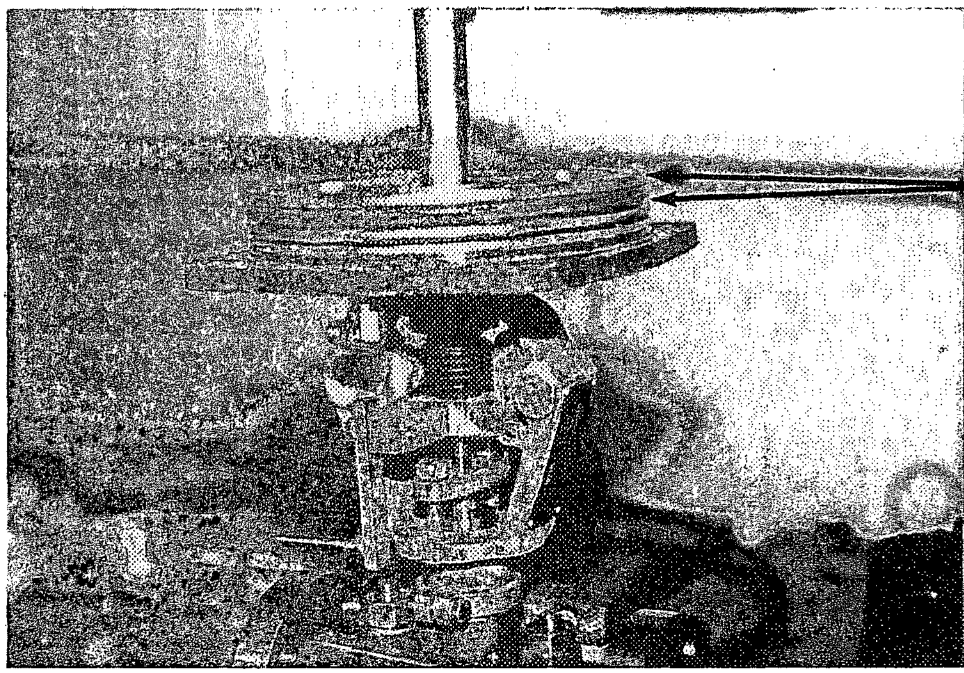



Figure 9.8 Annin actuator with cannister and spring removed

# 10 SAMPLING AND LABORATORY ANALYSIS

## 10.1 Sampling System

This section examines the design capability and operating practices of providing representative samples of favorable geometry quarantine tanks V-290 and V-291 to be utilized to ensure nuclear criticality safety. The uranium concentration in these quarantine tanks is restricted to 150 ppm of uranium before the contents may be transferred to the unfavorable geometry 20,000-gallon waste accumulation tank V-103. The quarantine tank's uranium concentration release limit is the primary criticality safety control mechanism for tank V-103.

### 10.1.1 General System Description

The aqueous waste (AW) quarantine tanks V-290 and V-291 collect AW from solvent-extraction Column A. AW normally contains metal impurities, nitric acid, and low levels of uranium. The AW quarantined in these tanks is sampled to determine its uranium content before it is transferred to the secondary nitrate waste collection tank (V-103) or to the solvent-extraction (SX) rework tank (V-225). The quarantine tanks are specifically designed as favorable geometry tanks. The maximum volume of each tank is 2270 liters (600 gallons); normally they were filled to 90 percent of their capacity. An AW transfer pump (P-290), common to both tanks, was used to recirculate each tank's contents before sampling and transfer. A sample line, connected to the recirculation system near the pump discharge point, routes a portion of the AW stream to the Uranium Recycle Unit (URU) laboratory where samples could be analyzed automatically for uranium and process contaminants (gadolinium, boron and cadmium) by one of two in-line inductively coupled argon plasma (ICAP) spectrophotometers or manually from a "grab" sample. Figure 10.1 shows the AW quarantine system and Figure 10.2 shows its associated sampling system.

### 10.1.2 Tank Recirculation Capabilities

Before the contents of tanks V-290 or V-291 are transferred to V-103 or V-225, a sample must be collected and analyzed for its uranium concentration. To ensure that analytical results are accurate, the sample must be representative of the actual tank contents at the time of sampling. One key component of representative sampling is to ensure that the fluid to be sampled is sufficiently mixed and homogeneous before the sample is collected. To ensure this condition, the tank must be (1) isolated to prevent the addition of further fluids and (2) completely mixed to ensure homogeneity.

The AW quarantine tanks are designed to be recirculated through transfer pump P-290, which takes suction from the bottom of the selected AW quarantine tank through 3-inch piping and discharges into the top of each tank through 1-inch piping. The 1-inch recirculation line also has an 0.3-inch diameter flow restricting orifice located between the pump discharge point and the tank inlet.

When the AW system is operated automatically, quarantine tank recirculation begins when the level reaches 50 percent. When the level reaches 90 percent of full, flow to the tank is automatically transferred to the empty tank. The procedure governing this process (PROD 103.02, Solvent-Extraction and Aqueous Waste Quarantine, Revision 3) specified that the full tank be recirculated an additional 15 minutes before sampling. A review of this procedure against the original system design and actual practices revealed the following:

1. The nominal flow rate from the SX system to the AW quarantine tanks was about 50 gallons per hour (gph). Prior to May 1987, this was the only source of the AW quarantine system. At 50 gph, the average fill rate of a quarantine tank to the 90-percent level (540 gallons) was about 10.8 hours. In May 1987, the process condensate (70 gph) and steam condensate (115 gph) from the head-end concentrator (described in Section 4.2.7.2) condenser and reboiler, respectively, were routed to the AW quarantine system. This addition increased the fill rate of a quarantine tank at the 90-percent level to about 2.3 hours. In November 1990, to reduce volume problems in the radwaste system, the licensee routed steam condensate (160 gph) from the uranyl nitrate product concentrator reboiler to the AW quarantine system. This addition made for a total 395 gallons of liquid being transferred to the AW quarantine system and a 90-percent fill rate of about 1.4 hours.

   The team also noted that the Facility Change Requests for the above modifications did not include an evaluation to determine what effect these additional volumes of liquid could have on the AW quarantine system, such as recirculation time for collecting a representative sample, or holding time between discharges during system upsets.

2. Although the tank contents were required to be recirculated for an additional 15 minutes after reaching the 90-percent full level, the team found no evidence that preoperational testing had been performed to determine that this recirculation time was adequate to ensure the homogeneity of a tank's contents prior to sampling. The specification sheet for the restricting orifice (FO-290) stated that the flow rate through the orifice was 23 liters per minute (6.1 gallons per minute) at 40 psia (actual flow rates could not be measured). Based on the rating of the orifice, a usable tank volume of 2043 liters (at the 90-percent level), and a 15-minute recirculation time, only about 17 percent of the tank volume could be recirculated after reaching the 90-percent level. Accepted industry practice recommends that a recirculation of at least two volumes (or testing to indicate otherwise) are needed to assure proper mixing. It is unlikely that representative samples could be collected from these tanks under existing operational conditions.

3. No direct-flow measuring device was installed in the recirculation system and several licensee operators and engineers interviewed were not aware of the normal pump discharge pressure when a tank was recirculating. During sample loop testing conducted after the incident, the team observed that the pressure indicator on the discharge side of pump P-290 was broken. Therefore, actual recirculation flow could not be determined.

4. The maximum recirculation flow through the AW quarantine tanks was calculated to be 23 liters per minute. With this low flow rate and the resulting laminar flow conditions in the tanks, it is possible that particulates could settle to the bottom of the tanks. Even with the slight slope of the tank bottoms, a low flow rate into the tanks may not create enough turbulence to keep particulates suspended or swept to the pump suction line at the low point in the tanks, and could lead to non-representative sampling of the tank.

5. A review of the process computer's historical data, URU control room operator tank-dump logs, chemistry laboratory logs, and other selected records indicated that before and during the incident on May 29, 1991, (1) samples were taken <u>prior</u> to the tanks being filled and isolated (as referenced in Table 6.1), and (2) in several cases, samples were not analyzed for their uranium concentration before being transferred to tanks V-103 or V-225. These practices are not indicative of representative sampling. Team interviews of URU operators and chemistry laboratory technicians confirmed that samples were routinely taken and analyzed when the AW quarantine tanks reached the 50-70-percent level to facilitate rapid transfers when the tank reached the 90-percent level.

### 10.1.3 Sampling from Tanks V-290 and V-291

A representative sample of the contents of tank V-290 or V-291 must be collected to ensure accurate analytical results. If the tanks are sufficiently isolated and recirculated before sampling, then a representative aliquot can be collected from the tanks and delivered to the sample point. Another key component of representative sampling in this case would be to ensure that (1) sampling lines are sufficiently purged prior to sample collection and (2) conditions for isokinetic sampling are met if particulates are present in the sample stream (especially during upset conditions).

Samples were collected during the recirculation of tank contents through a 0.5-inch, 304L stainless steel, Schedule-40 sample line. This line was connected to the 1-inch recirculation piping near the discharge point for pump P-290 and runs approximately 110 feet to the URU laboratory where it passes through a valve cabinet containing an in-line filter, flow sensing valve, pressure indicator, and other associated valving. The line then passes in series through two ICAP automatic analyzers and near a sample hood where a 0.25-inch wide, 80-inch long grab sampling line is connected. This hood also contains grab sampling points for other plant processes. The line then passes through a flow-restricting orifice (0.125-inch diameter) and back down to the 3-inch inlet piping of P-290 near the pump (Figures 10.1

and 10.2). The pressure differential across P-290 results in flow through the sample loop. Since the pressure regulator in the system had been removed due to operability problems, the sample loop in the URU laboratory was being controlled manually to about 10 psi so as not to damage sensitive multiport valves associated with the ICAP analyzers. After the incident, testing indicated that flow through the loop within the normal operating pressure was approximately 1 gallon per minute, as per design. It should be noted that a sample loop flow switch had also been out of service for a considerable time period.

It is important to note that the sample line from P-290 to the URU laboratory contained many changes in flow direction, none of which were long radius bends. In fact, all turns were angled 90 degrees with standard "L" (square elbow) fittings. If particulates were present in the sample, the possibility existed for them to be deposited at bends and along straight runs of piping. The amount of deposition of any insoluble materials, especially any undissolved uranium, is dependent on many factors, including particle size distribution, fluid density and viscosity, and flow rate through the sample piping. If there is any two-phase flow in the sample piping (aqueous and organic mixtures), it could cause increased deposition or even a "crud burst" in the sample piping, depending on processing conditions at the time. The resulting samples would not be representative of actual tank conditions, especially during upset conditions.

The licensee informed the team that a significant amount of solvent from the extraction columns passed through tank T-292 and into tank V-291, and possibly V-290, during the incident. This resulted in a significant volume of solvent collecting on top of the aqueous solution in the tank, which was removed by the licensee. Since recirculation suction is taken from the bottom of the tank, it is possible that (1) this upset condition, such as solvent laden with high concentrations of uranium, could not be readily detected through the sampling system; and (2) if some mixing were to occur within the tank, two-phase flow through the sampling line could drastically affect the representativeness of the sample for the same reasons discussed above.

Discussions with laboratory personnel also indicated that it was not unusual to collect small amounts of solvent along with aqueous samples from the laboratory sample point for tanks V-290 and V-291. These comments were validated during a sample loop flow test conducted on June 10, 1991 (see Section 10.2), when a small amount of solvent appeared to be evident in a sample collected at the end of the loop, most likely as a result of the incident.

Approximately two weeks before the incident, manual sampling of tanks V-290 and V-291 was initiated in the URU laboratory hood because of problems with the automatic ICAP analyzers. The manual sample point is from a 0.25-inch line, 80 inches long, connected to the main 0.5-inch sample loop. ASTM Standard D3370-82, Standard Practices for Sampling Water, recommends minimum purge times for grab sample lines. According to the standard, at a flow rate of 500 ml/minute, a 0.25-inch sample line should be purged for 10 seconds per foot of line. Testing done after the incident resulted in a flow rate of about 1.6 liters per minute (l/min) out of this line. With a flow rate of 1.6 l/min, the 80-inch sample line should be purged for about 1 minute. Although not specified by procedure, lab analysts indicated that this sample line, along with other sample lines located in the sample hood, was usually purged for only about 15 or 20 seconds prior to sample collection, a period which clearly was not adequate.

As stated earlier, the sample line to the URU laboratory is routed from the recirculation line near the discharge point of pump P-290 to the laboratory and back to the 3-inch inlet line near the suction of pump P-290 (Figure 10.1). If the recirculation line becomes blocked at the orifice past the sample line inlet, P-290 would pump fluid in a closed loop to the URU laboratory and back to the pump suction. Thus, recirculation flow from the in-service AW quarantine tank would cease. With no flow measurement capability in the recirculation line, there would be no obvious way to detect this abnormal condition and samples could be collected and analyzed with plant personnel being unaware that the tank was not being properly recirculated.

### 10.1.4 Sample Loop Purging

ASTM D-3370-82, Standard Practices for Sampling Water, suggests using purging times to provide three to four times the sample line volume to ensure representativeness. Testing of the AW quarantine tanks sample loop conducted on June 10, 1991 (Section 10.2) verified the design flow rate of approximately 4 liters per minute. The volume of the sample loop was calculated to be approximately 6.7 liters. If the AW quarantine tanks were recirculated for 15 minutes as stated in the procedure prior to sampling, nine sample line volumes would be purged through the loop. Under normal system conditions, this purge time would be adequate.

The test results also indicated that if the sample return line became plugged, then the lab grab sample point would have to purge for approximately 4.2 minutes before a representative sample would arrive from the quarantine tanks. With a flow rate of 1.6 liters per minute out of the sample point, only one sample line volume would be purged through the loop prior to sample collection. It is unlikely that this purge time would provide a representative sample to the grab sample point under plugged conditions.

### 10.1.5 Sampling System Pre-operational Testing

Although design considerations in a sampling system are important in ensuring that representative samples are collected from various points in the process, only pre- and post-operational testing can verify that tank recirculation times are adequate and that samples collected at various points are indeed representative. No evidence was presented during the investigation to show that this type of testing was ever attempted.

### 10.1.6 Tank T-292 Sampling

The AW recirculation tank (T-292) is an intermediate favorable geometry tank of between the solvent-extraction columns and tanks V-290 and V-291. Pump P-292 recirculates aqueous wastes through a heat exchanger, E-292. A sample line located near the pump discharge routes a sample to the URU laboratory and back, similar to the process for the AW/Surge-tanks sample line. The use of this line was discontinued in 1987 when a leak developed at a pipe weld. A process engineer indicated that this line was never repaired because a more immediate sample could be collected to determine uranium loss from solvent extraction from a point near the bottom of solvent-extraction Column A. A grab sample could be collected locally from a point near the discharge of pump P-292, however; purportedly it was not used.

## 10.2 Aqueous Waste Quarantine Tanks' Sample Loop Flow Test

As part of this investigation, the licensee performed a functional test of the aqueous waste quarantine tanks' sampling system loop (Figure 10.2) on June 10, 1991. The purpose of the test was to determine if there were any restrictions in the sampling loop that may have contributed to non-representative sampling of the contents of aqueous waste quarantine tanks during the incident. The sample loop had not been in use since the shutdown of the solvent-extraction process on May 29, 1991. The scope of the test included evaluating the present status of the sampling loop by (1) measuring the flow rate through the sampling loop; (2) sampling the selected aqueous waste quarantine tanks simultaneously at two locations (the URU sample hood and the suction side of quarantine tanks' recirculation pump P-290); (3) looking for foreign material in samples collected, (4) disassembling and inspecting certain components of the sampling system for blockage, and (5) conducting a laboratory analysis of selected samples (Table 10.1). A summary of observations made during the test is as follows:

1:30 p.m. With the system shutdown, operators opened the sample valve on the inlet of P-290 and collected about 600 milliliters (ml) of liquid.

The liquid collected appeared somewhat clear and no foreign material was observed.

2:32 p.m. Operators recirculated the contents of QT V-291 for 15 minutes with the tank 80 percent full.

2:50 p.m. They opened the URU lab manual sample valve until a steady flow was obtained and inspected the liquid collected for clarity and foreign materials.

They observed no foreign material in the liquid. Sample line pressure fluctuated from 12 to 18 psi and leveled off at about 12 psi.

2:52 p.m. They collected a 30-second sample from the manual sample valve in the URU laboratory sample hood and calculated the sample flow rate.

About 835 ml of the sample was collected (in 30 seconds), which was equivalent to a flow rate of 1.67 l/min. The sample collected also appeared to have a very light gray tint, and appeared to have a very minute amount of solvent and specks of foreign material floating on the surface.

3:00 p.m. They drew simultaneous samples from the URU lab's manual sample station and at the sample line on the suction side of pump P-290.

The sample obtained at the inlet side of P-290 was distinctively darker in color than the sample obtained in the URU lab (Figure 10.3).

3:10 p.m. With the URU lab AW quarantine manual sample valve open, they closed the down stream sample loop's return valve to observe line pressure (22 psi). With the return valve closed, the manual sample valve was closed to observe line pressure (24 psi).

3:15 p.m. They disconnected the sample return line at the isolation valve to the inlet side of pump P-290 and collected a 1-minute sample.

Approximately four liters of liquid were collected, indicating a flow rate of just over one gallon per minute (system design flow). The liquid collected also appeared to be darker than the sample previously obtained in the URU lab.

3:30 p.m. They removed and inspected the sample loop's in-line filter (a stainless steel screen of about 60-90 mesh) located upstream of the manual sample valve in the URU lab.

They observed a small amount of humus-like material that had collected on the filter (Figure 10.4).

4:15 p.m. They disassembled and inspected the sample loop's flow orifice (0.125 inch), located in the ceiling of the URU lab and on a horizonal plane down stream of the manual sampling line tie-in. During the disassembly, liquid drained from the line was collected in a clean pan for inspection along with the orifice.

The orifice was observed to be free of any foreign debris; however, several metallic fragments (Figure 10.5) and other debris (having the appearance of corrosion products) were collected with the liquid that drained from the sample loop during disassembly of the orifice. Because of the size of the fragments (Figure 10.5), it was assumed that they had come from the upstream side of the orifice. Coloration on the surface plate of the orifice made it appear that the corrosion products could have been dammed up against the surface plate below the orifice opening.

The above tests and observations did not identify a blockage in the sampling loop. The flow rate of 4.0 l/m was consistent with design flow of 1.0 gpm at 10 psi. The difference in the clarity between the two simultaneous samples appeared indicative of an inadequate circulation time (15 minutes) of the quarantine tanks as to provide a homogenous mix of the tank contents before this sample was collected. The metal fragments were large enough to restrict flow through the sample loop if they impinged on the opening of the orifice. The amount of material observed on the filter would not appear to have a significant impact on the flow rate through the sample loop.

## 10.3 Uranium Recycle Unit (URU) Laboratory Analysis

The URU laboratory was designed and equipped to provide analytical data to verify that various waste processes were operating within specified parameters. To this end, the laboratory measured uranium and other chemical concentrations in the various wastes in-process and waste discharge streams, in addition to measuring and verifying uranium and impurity concentrations in uranyl nitrate.

As part of the original laboratory design, the various process and waste streams were sampled through continuous-flow sampling loops that provided samples to either of two ICAPs (the term used for the older instruments). The ICAPs automatically analyzed samples by an automatic sample selection system and reported the data to the process control system, to the laboratory chemist (Lab Operator), and to a laboratory host computer.

Prior to May 15, 1991, the ICAPs were connected to the aqueous waste quarantine tank (AWQT) sampling system, which continuously recirculated aqueous waste to a sampling cell (of about 250 milliliters), which the ICAPs automatically sampled by instrument demand. This sample loop contains a 60 to 90-micron porous metal filter to remove large particles to prevent blockage and possible damage to the ICAPs. The AWQT sampling system was also equipped with a manually operated sample valve, located down stream of the ICAPs, that could be used when problems occurred with the automatic sampling system. Because of problems with the automatic sampling mode (leaks due to corrosion and instrument failure), the licensee initiated routine manual sampling of the AWQTs around May 15, 1991.

The licensee was in the process of fabricating a new sampling cabinet and equipping the URU lab with new ICPs because of problems with the ICAP's automatic sample selection valves and the instruments themselves. Although the new sampling system had not been introduced to the lab, two new ICPs had been installed. The new ICPs were equipped with their own computer and were in the process of being qualified for use. No procedures had been developed for analyzing samples on the new ICPs at the time of the incident.

### 10.3.1 URU Laboratory Quality Control

The URU laboratory had no specific quality control/quality assurance procedures. The licensee considered the URU lab as a process control laboratory and treated sample results on a go-no-go basis for control purposes. Instrument response checks using standards of 20 and 40 ppm of uranium were run with each batch of samples and had to agree within $\pm$2.5 ppm; if not, they were re-analyzed for verification. The licensee's procedures delineated that if the standard results were not within $\pm$2.5 ppm, the instrument must be recalibrated. The request for calibration came from the computer. Records of these calibrations were not available. The uranium calibration check standards are made up by the licensee's Standards Laboratory. This standards procedure preparation apparently has not been updated to show the current 20 ppm and 40 ppm standards. When present in large quantities, calcium and iron interfere with sample analysis. Although there was a constant calcium background value programmed in the computer, it may not have the appropriate correction bias to account for the calcium present in a given sample.

For ICAP samples, the results were purportedly relatively accurate ($\pm$25 percent) with a higher detection limit of about 15 to 20 ppm of uranium compared with the 1 ppm of uranium or better with the new ICPs. However, no documentation existed to corroborate the accuracy of the ICAP analyses.

As part of a continuing evaluation of the Uranium Recovery Unit lab's analytical capabilities, on June 26, 1991, the Incident Investigation Team consultant observed a lab analyst prepare and analyze two samples using the new ICPs. The samples were from the Rad Waste and Nitrate Waste streams, one of which was slightly cloudy but cleared after the addition of acid. The samples were prepared in accordance with established procedures. Uranium standard solutions of 20 and 40 ppm of uranium were analyzed prior to analyzing the prepared samples. The standards agreed to within $\pm$2.5 ppm and the sample results for both samples were less than the go-no-go 25 ppm of uranium, which was considered within the expected range for these samples. The analyst appeared to use good laboratory techniques.

The precision and bias of the new ICPs were noted to be $\pm$10 percent. The bias was noted as 6.45 percent and 3.82 percent and Relative Standard Deviation as 2.83 percent and 2.04 percent. This range was noted to be consistent with Tables 3 and 4 in the ASTM Standard C1111-88, "Standard Elements in Waste Streams by Inductively Coupled Plasma-Atomic Emission Spectroscopy."

The IIT consultant noted that one of the analytical steps was not in conformance with Section 10.2 of ASTM C1111. Specifically, filtering solids and discarding the precipitate could lead to non-conservative sample results.

### 10.3.2 Sample Results During the Incident

On the evening of May 28 through 5:30 a.m. on May 29, 1991, the computer for the ICAPs was down and the AWQT samples were apparently being analyzed on the new ICPs. The new ICPs had not been fully qualified for official use nor had their operating procedures been developed; however, the licensee believed that the sample results would have been accurate to within approximately $\pm 25$ percent of the actual sample concentration.

During the incident when the interface control was lost in the solvent-extraction A Column, organic solvent passed to the AW stream, undoubtedly flooded and overwhelmed the Decanter, and passed to the quarantine tanks. The design of the sampling system probably precluded any significant volume of solvent from passing into the sample lines. The licensee's process engineer concurred with the team in this assessment. No samples were available to examine for an organic layer during the team's evaluation. On June 10, 1991, the team observed that the V-103 composite sample did not appear to contain an organic layer. The process engineer stated that on interviewing the analyst on shift at the time, the samples only appeared cloudy and without any discernible organic layer. The Laboratory Technical Analyst stated that if organic matter had been in the sample, it would have damaged the ICP. It appears very plausible that the difference in the material balance numbers based on the solvent-extraction feed to the A Column and the integrated uranium concentration in the various tank 290 and 291 dumps is accounted for by the organic solvent which contained the bulk of the uranium since it was not picked up to the aqueous phase sample. It is not plausible for the ICP analysis error to be of sufficient magnitude to account for this very large difference (the estimated adjusted value of 12,000 ppm of uranium determined by the licensee versus the 40-85 ppm of uranium range observed for samples that were measured during the event). The discrepancy must have been caused by non-representative sampling.

Although the URU lab had a procedure for preparing and analyzing non-routine samples, and that would apply to manual samples, the procedure was very general and would not apply to the new ICPs. The procedure for non-routine samples depended on what type of analysis was requested and required a significant amount of judgement in its use. Based on the review of sample results of AW quarantine tanks V-290 and V-291, the V-104 tank at the Waste Treatment Facility, and interviews with the URU lab's chemistry staff, the team found that inadequate attention was given to the samples being analyzed. Specifically, because of the upset conditions that existed during the incident it is apparent that samples would have contained two phases, liquid and solids. Without adequate procedures and proper treatment of the samples (i.e., the addition of acid), the sample measurements would not have been representative of the samples being analyzed. As observed by NRC personnel immediately following the event, poor laboratory techniques were being used in the URU laboratory in the preparation of samples for analysis. For example:

> Good laboratory techniques demand that samples taken from tank V-104 should have been placed in a beaker and with a magnetic stirrer, stirred for the same amount of time, and with the magnetic stirrer plate at the same setting. While stirring the sample, to take an aliquot the pipet should have been inserted to the same depth for each sample.[1]

This recommended technique was not observed in the preparation of the sample from tank V-104 following the incident. Similar poor techniques were also observed at other laboratories by NRC personnel during the recovery phase. These poor techniques led to inaccurate sample measurements and an inadequate assessment of uranium concentrations. A detailed specific procedure for analyzing aqueous waste samples manually did not exist. However, a new interim procedure was developed and issued on May 31, 1991, following the incident. This new procedure provided adequate guidance and techniques to assure that sample results were more representative of the sample collected.

Immediately following the incident, the licensee had difficulty determining the uranium concentration in tank V-104. Tank V-104 has three sample points (A, C, and D). Point D samples the bottom (cone) portion of the tank, point C samples the lower portion of the tank above the cone, and sample point A (located below the mid point of the tank) samples the upper tank level from a 4-inch standpipe within the tank. From interviews with the licensee's staff and the NRC recovery team, the team determined that the operator was apparently unaware of the configuration of sample point A and believed that he was collecting a sample from near the middle of the tank, and not from the surface of the tank's contents. The liquid at the surface would have not been representative of the tank's contents and would have contained floating debris. Thus, an analysis of samples collected from this point would provide erroneous results, especially if the sample did not receive special treatment before being analyzed. The licensee obtained more consistent sample results after instituting a "dip" sampling method, which drew material from well within the tank.

## 10.4 Comparison of Aqueous Waste Daily Composite Samples

Automatic composite samples of the AW feed stream to tank V-103 were collected at about 8:00 a.m. each day. Figure 10.6 shows the comparison of the daily composite sample measurements (in ppm of uranium) and the average uranium concentrations of tanks V-290 and V-291 for the same time period. The daily composite samples were analyzed in the Environmental Laboratory, while the aqueous waste quarantine tank samples were analyzed in the URU lab. The comparison indicates that the composite sample results were consistently higher than the average of the sample results observed in the URU laboratory for quarantine tanks V-290 and V-291. This difference could be indicative of (1) inadequate laboratory

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[1] Memorandum from L. A. Roche, Office of Nuclear Materials Safety and Safeguards, U.S. NRC, to R. Scarano, Team Leader, Incident Investigation Team, GE Fuel Plant. Subject: Investigation of Potential Criticality Problem at General Electric (GE) Fuel Plant at Wilmington, North Carolina, June 17, 1991.

measurement, (2) non-representative sampling of the AW quarantine tanks (circulation time of the QTs or sampling loop design/perturbations), (3) or calibration problem of the composite sampling system.

## 10.5 Uranium Mass Balance of Incident

On June 6, 1991, the licensee presented the team with their evaluation of the mass balance of uranium accumulated at various locations during the incident. The licensee estimated that 150 kgs of uranium were transferred to tank V-103 and ultimately to tank V-104 based on a comparison of the difference between the weight in kilograms of uranium input to the SX system and the kilograms of uranium that could be accounted for in SX-related tanks in the URU facility. Although the licensee's evaluation included assumptions because of the lack of sample measurements of the AW quarantine tanks during the incident, the team did not have any specific concerns relative to the licensee's evaluation. The team independently estimated the expected loss and assumed that during the interval from May 28, 1991, 8:30 p.m., through May 29, 1991, 5:00 a.m. (when the SX system was ineffective), that the uranium input stream was going directly to the AW system. From computer historical data, the team could determine that the AW discharge valve from the SX system was open for about 4.12 hours during the incident. Assuming average values for the feed and feed rate, and assuming that none of the AW sample measurements were representative, the team's estimate, below, was in relative agreement with the licensee's:

Average SX feed = 0.18 kgs U/liter
Average SX feed rate = 200 liters per hour
Approximate total time of upset conditions = 4.12 hours
<u>(accounting for the time that the SX Column A manual valve was closed)</u>

0.18 kgs U/liter x 200 liters per hour x 4.12 hours = **148 kgs U**

Table 10.1 URU sample loop flow test sample analysis*

| Sample | URU Lab #1 ICP | Chemet Lab | (Old ICP) URU-ICAP 1 |
|---|---|---|---|
| A | 27.4** | 42.0 | |
| B | 76.5 | 79.85 | 100 |
| C | 43.5 | 44.6 | 49.7 |
| D | 40.4 | 41.3 | |
| E | 43.7 | 41.2 | 50.9 |
| F | 5976 | 6976 | |
| F1 | 5739 | 6420 | |

*All results are in parts per million.
**Particles were plugging nebulizer (two nebulizers were destroyed).

Identification

A 1st Liquid Out of Sample Loop
B AQW, 2:30 p.m., 6/10, Sample Line Loop Test
C V-291, 6/10, 3:00 p.m., Simultaneous P-250 inlet
D AQW, 3:00 p.m., 6/10, Simultaneous URU Lab
E Lab Loop Return, 4:00 p.m., 6/10
F AQ Comp 5/29 Solution Only
F1 AQ Comp 5/29 Solution, Particle, acid mixed

## 10.6 Findings and Conclusions

Based on the foregoing analysis, the team's findings relative to the sampling of the aqueous waste quarantine tanks are as follows:

- During the incident, many tanks were sampled before being completely filled and isolated, which would preclude representative sampling. In addition, several tanks of AW were not analyzed prior to transferring their contents to the unfavorable geometry tank V-103.

- A flow orifice (FO-290) installed in the recirculation line was so small that flow would be impeded to the extent that proper mixing of tank contents could not be achieved prior to sampling. With the tank recirculation flow rates as estimated, a recirculation time of 15 minutes was inadequate because it allowed recirculation of only 17 percent of tank volume after it reached the 90-percent fill level.

- The 110-foot long sampling line from the quarantine tanks to the URU laboratory had numerous 90-degree bends where particulates could be deposited.

- Low recirculation flow rates could result in particulates settling to the bottom of the AW quarantine tanks, which would preclude these particulates from being collected when the tank was sampled.

- The sampling system was not designed for two-phase sampling, which can occur during upset conditions. Representative sampling and analysis are most important during upset conditions.

- The possibility of two-phase flow in the sample line, especially during upset conditions or during the end of tank transfers, could greatly increase the likelihood of particulate deposition and nonrepresentative sampling.

- The laboratory grab sample point was normally not purged adequately prior to sample collection.

- Blockage in the AW quarantine tank recirculation lines could result in closed loop flow through the laboratory sample loop with plant personnel being unaware of the condition.

- Testing was never done to verify the adequacy of tank recirculation times and sampling capabilities.

- No evaluation was performed to determine what effect additional inputs of liquids could have on the AW operating procedures.

In summary, the sampling system's deficient design would make it highly improbable that samples were representative of tank contents. In addition, samples were routinely taken before the AW quarantine tanks were filled, further exacerbating the acquisition of non-representative sampling.

A review of URU laboratory practices indicated that:

- The first sample collected from the manual sample point in the URU lab, after 15 minutes of quarantine tank recirculation, indicated a value of about 50 percent higher than the sample collected about 30 minute later (simultaneous P-290 and URU lab samples). Although not conclusive, this difference could be an indication of inadequate tank circulation time and/or purging of the entire sample loop.

- The sample results from the new ICP in the URU lab were in relative agreement with the sample results from the Chemet lab.

- The sample results from the URU lab ICAP were about 20 percent higher than those from the new ICP and from the Chemet lab sample results. This difference coincides with the observations delineated in Section 10.3, and is within the relative range of $\pm$25 percent of true sample concentration.

- A review of the URU laboratory process indicated that the new ICPs are state-of-the-art instruments and their use to analyze uranium in waste streams is consistent with standard nuclear industry practices. This method is also used to qualify the high-level waste (HLW vitrified into a borosilicate glass waste form) at the Department of Energy waste sites throughout the country.

  The ICAPs are somewhat less sensitive and were not in use at the time of this review; however, the results produced by this instrument would be expected to be adequate ($\pm$25 percent) at the time of the incident.

- The procedures used at the time of the incident contained no estimates of the precision and bias standards for the instruments used.

- The overall practices used in the URU lab appeared adequate for process control purposes. However, samples which had nuclear safety implications were not processed using quality control in procedures and lacked records maintenance control, as specified in American Society for Testing and Materials standard C1009.

- The licensee did not establish limiting conditions of operations and did not review and rigidly apply criticality safety limits for uranium values.

- The sample results obtained during and after the incident appear to be relatively accurate and consistent with those expected of the instruments used. The degree to which the sample results reflected true concentrations would depend on whether the sample was representative of the component being sampled.

The comparison of aqueous waste daily composite samples indicates that the composite sample results were consistently higher than the average of the sample results observed in the URU laboratory for quarantine tanks V-290 and V-291. This difference could be indicative of (1) inadequate laboratory measurement, (2) non-representative sampling of the AW quarantine tanks (circulation time of the QTs or sampling loop design/perturbations), (3) or calibration problem of the composite sampling system.

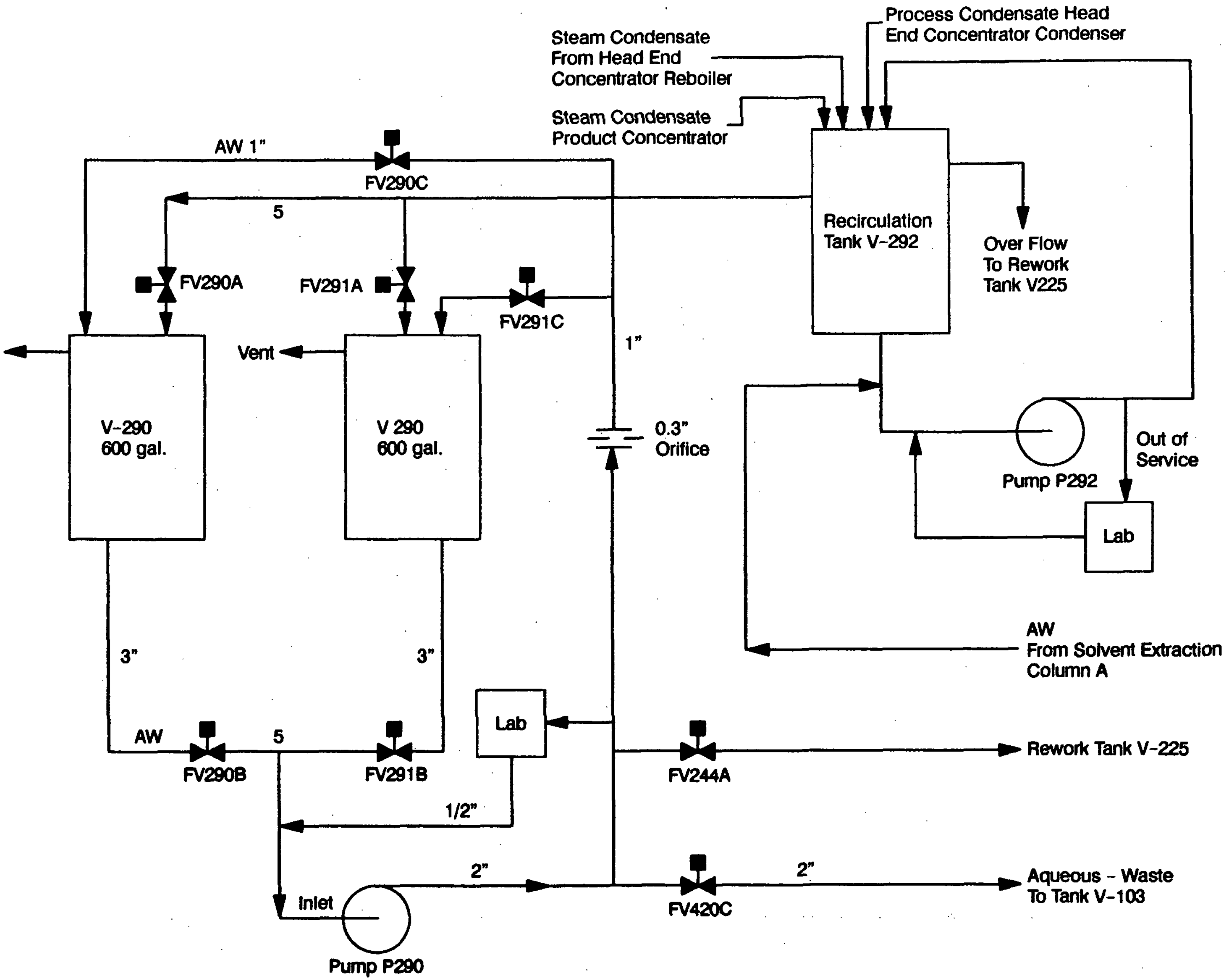



Figure 10.1 Aqueous waste (AW) quarantine

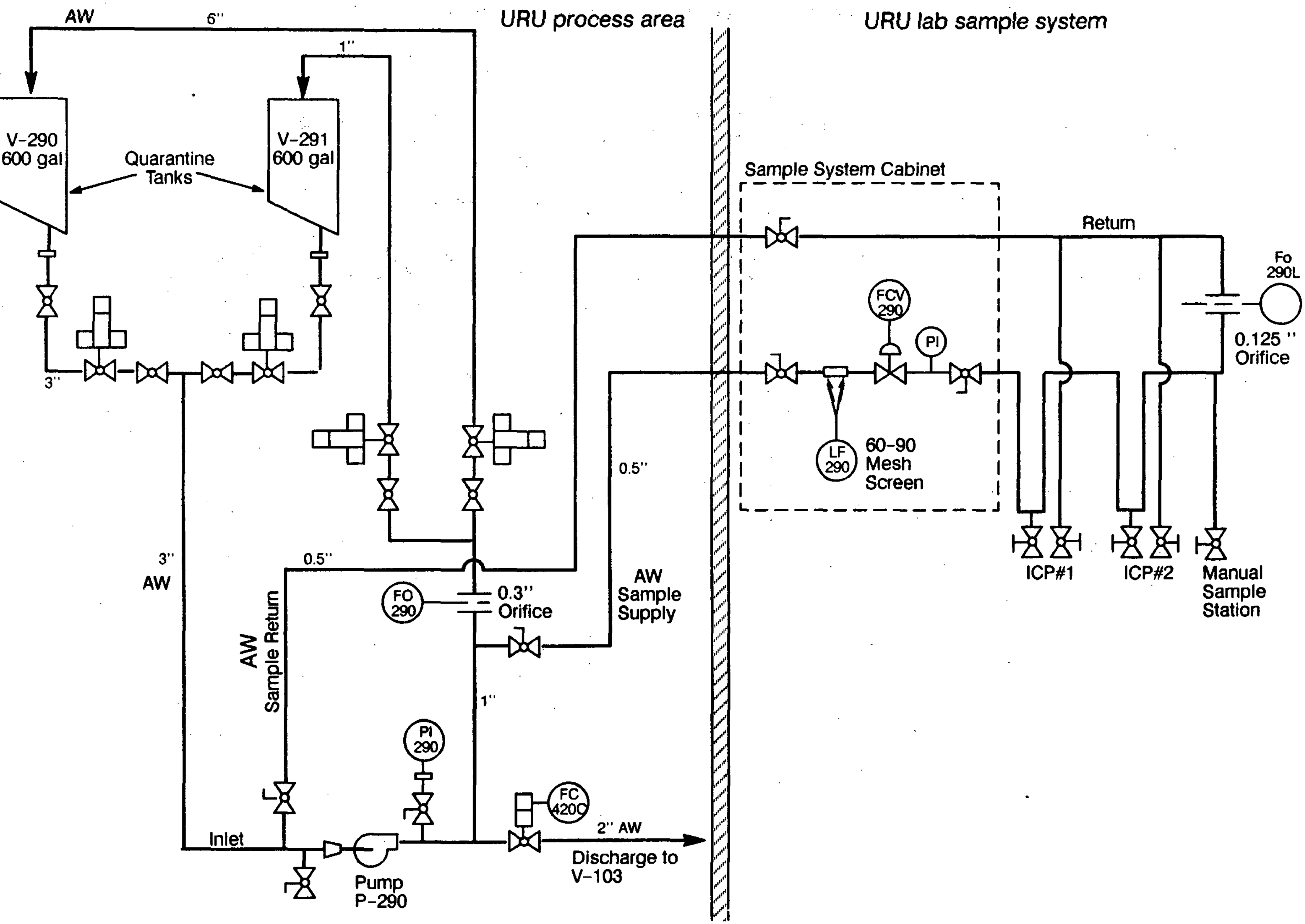
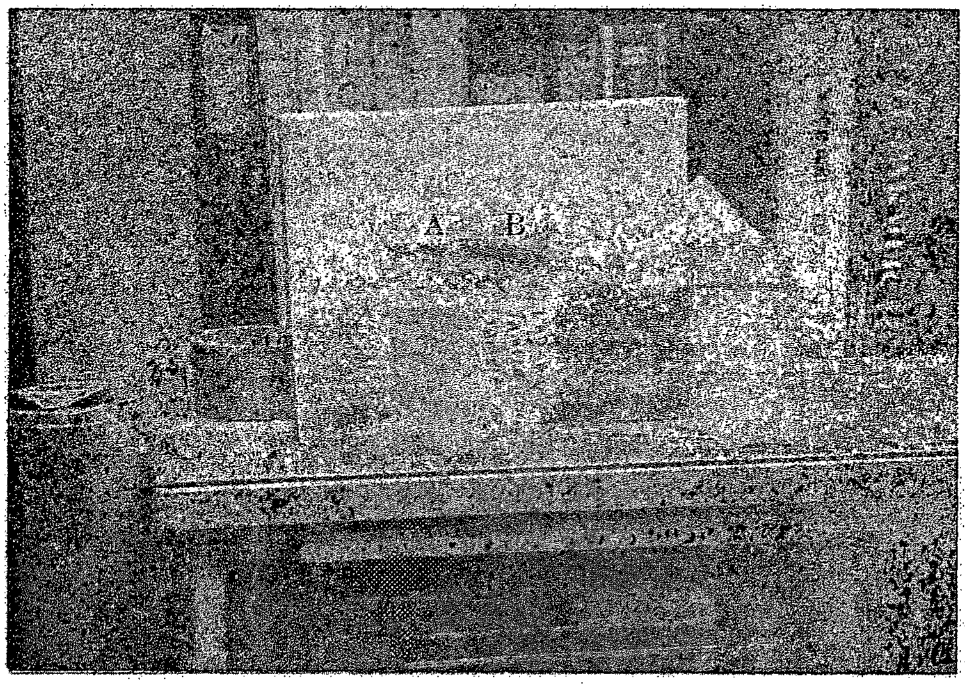



Figure 10.2 Aqueous waste (AW) quarantine tank sampling system




A - Pump (p-290) Inlet
B - URU Lab

Figure 10.3 Simultaneous aqueous waste samples from tank V-291 (6/10/91, 3:00 p.m.)

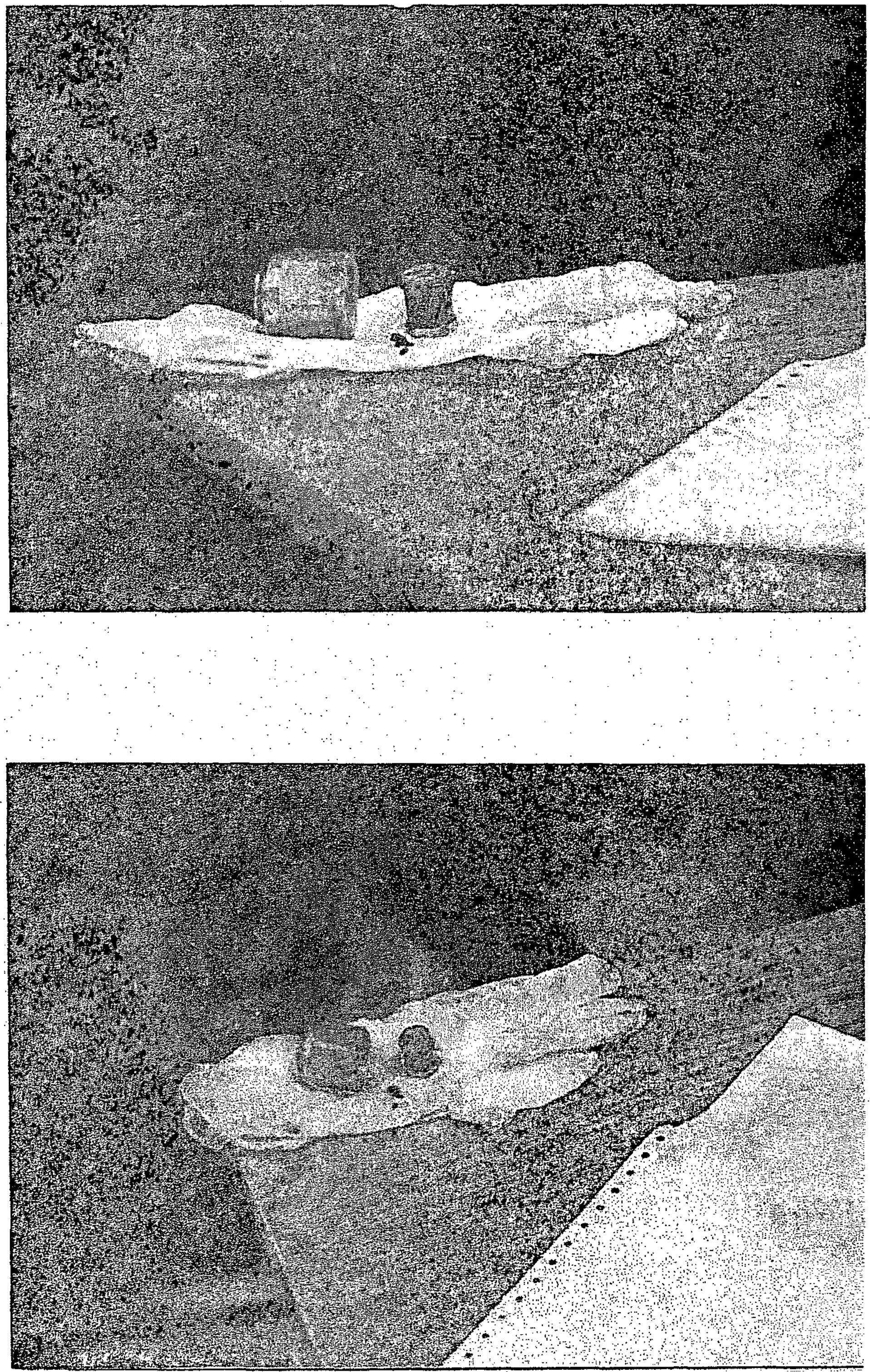


Figure 10.4 AW sampling system 60 – 90 mesh filter

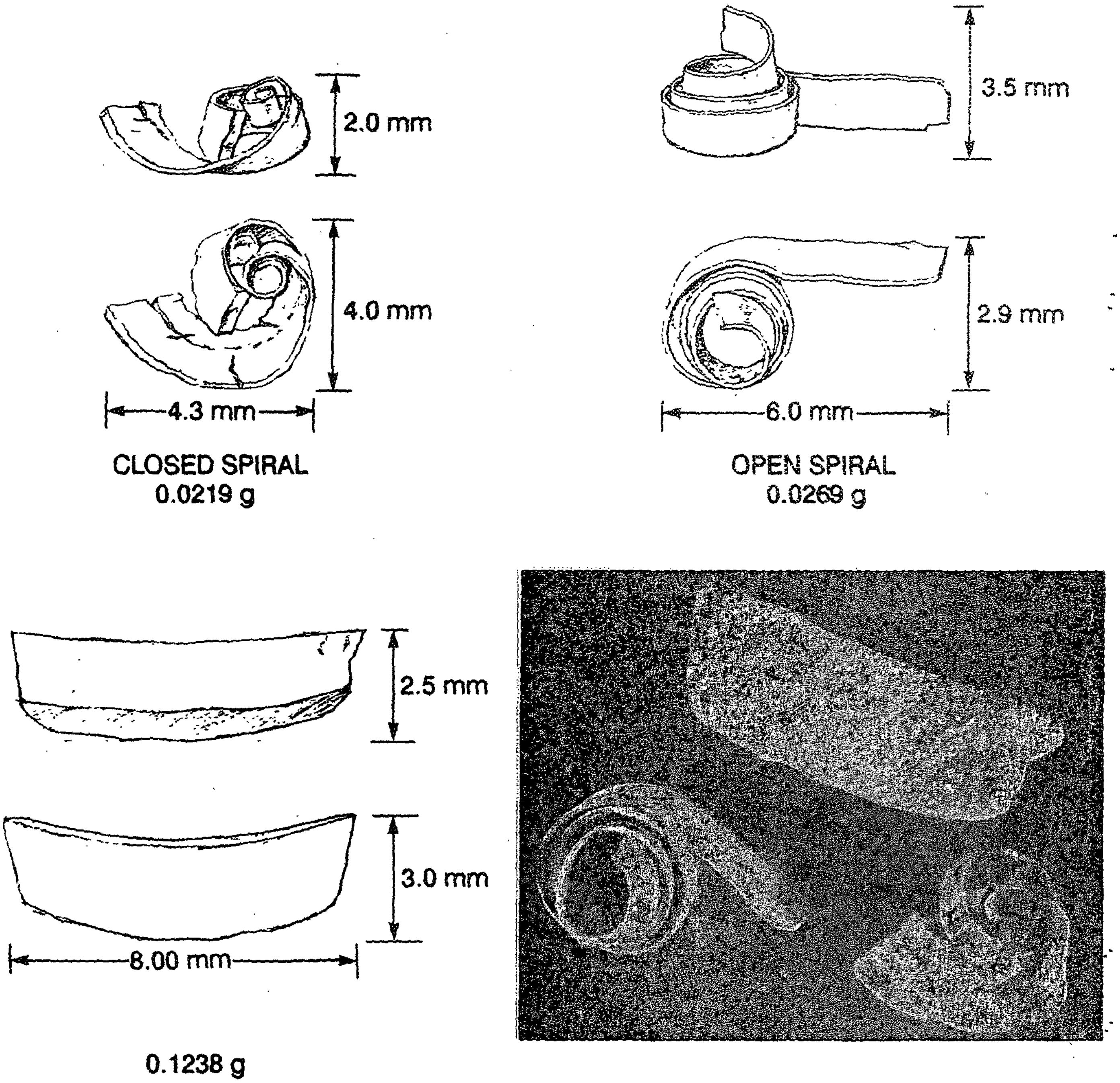



Figure 10.5 Metal fragments found near FO-290L

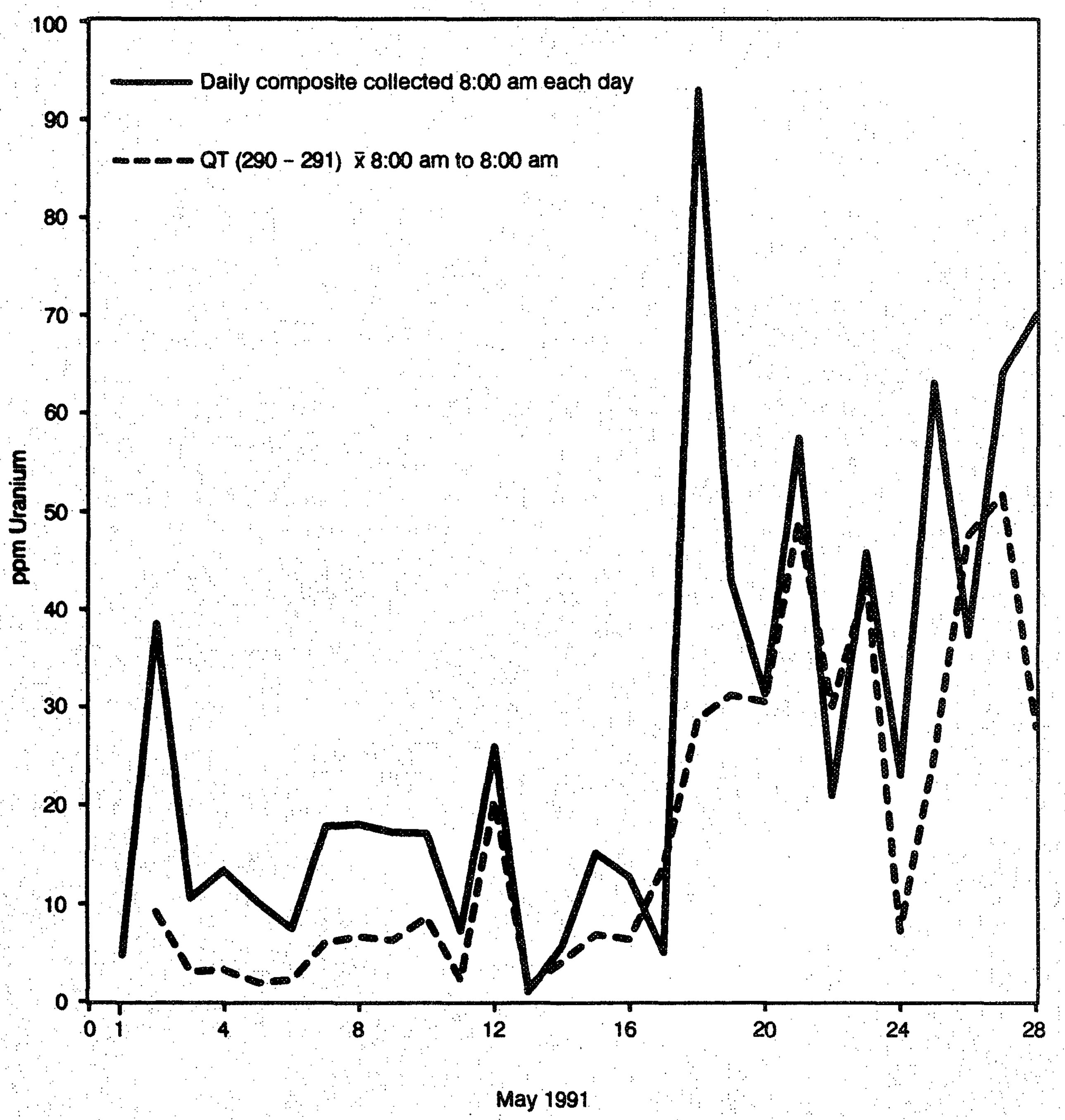



Figure 10.6 Aqueous waste composite and quarantine samples

# 11 PRECURSORS

## 11.1 Background

The incident at the General Electric (GE) Nuclear Fuels and Component Manufacturing Operation was neither unique nor without precedent because incidents involving high concentrations of enriched uranium in unfavorable geometry configurations have previously occurred in the industry. Some of these incidents have actually resulted in criticality accidents while others have had a very high potential for a criticality. The eight previous incidents where nuclear criticalities occurred in fuel cycle facilities involved operations with either highly enriched uranium or plutonium (Table 11.1). Of these eight incidents, three occurred when highly enriched uranium was transferred to unfavorable geometry vessels. Although a nuclear criticality accident has not yet occurred at a low enriched uranium fuel cycle facility, one could occur unless the appropriate controls discussed in Section 7 are in place. Precursor incidents involving high concentrations of enriched uranium at other plants have occurred twice since 1989. Although these events involved highly enriched uranium and did not result in a nuclear criticality accident, they shared some of the causes implicated in the May 29, 1991, incident at the GE-Wilmington facility: inadequate management attention to nuclear criticality safety programs, lack of mandatory and enforced procedural compliance, inadequate training programs, and the need for additional monitoring devices. A description of the 1989 and 1990 industry events and NRC's notifications to industry about them are found in Section 11.2.

Related and precursor incidents at GE (the licensee) are listed in Table 11.2 and are described in detail in Section 11.3.

## 11.2 Operating Experience Information Issued by NRC

This section describes operating experience information issued by the U.S. Nuclear Regulatory Commission (NRC), in the form of Information Notices (IN) that are applicable to the incident that occurred on May 29, 1991, at the GE Nuclear Fuel and Component Manufacturing Operation Facility. INs provide information to NRC-licensed facilities about potentially significant problems or operational events that are relevant to safety, safeguards, or environmental issues. Licensees are expected to review the information included for its applicability to their facilities and consider action, as appropriate, to avoid similar problems or events. However, suggestions contained in INs do not constitute NRC requirements; therefore, NRC requires no specific action be taken or written response be made after receipt of an IN.

# Table 11.1 Criticality Accidents in Fuel Cycle Facilities

| Date | Location | Process | Cause | Total Fissions | Duration | Personnel Exposures | Contamination In-plant | Contamination Out of Building |
|---|---|---|---|---|---|---|---|---|
| 06/16/58 | Y-12 Processing Plant, Oak Ridge, TN | Recovery of highly enriched uranium by chemical methods. | Wash water added to $UO_2(NO_3)_2$ solution in 55-gal. drum. | $1.3 \times 10^{18}$ | 18 min | 8 people. Doses of 461, 418, 413, 341, 298, 87, 29 rads. No fatalities. | Small local contamination. | None reported. |
| 12/30/58 | Pu Processing Plant, Los Alamos, NM | Recovery of plutonium from scrap. | Liquid phases of plutonium separated out. | $1.5 \times 10^{17}$ | 1 sec | 3 people. Doses of 12,000, 134, 53 rads. One fatality. | None reported. | None reported. |
| 10/16/59 | Idaho Chemical Processing Plant, Idaho Reactor Test Site | Transfer of highly enriched uranium solution. | Solution transferred to unsafe geometry 5000-gal tank. | $4 \times 10^{19}$ | 15 to 20 min | 19 people. No direct gamma or neutron dose because tank was shielded, but beta doses from releases of 50 rem, 32 rads, and smaller amounts for 17 other people. | Yes. Airborne beta activity. | Not reported. |
| 01/25/61 | Idaho Chemical Processing Plant, Idaho Reactor Test Site | Transfer of highly enriched uranium solution. | Solution transferred to unsafe geometry tank. | $6 \times 10^{17}$ | 1 sec | None. Shielded operation. | None reported. | None reported. |
| 04/07/62 | Hanford Works, Richland, WA | Plutonium processing. | Plutonium solution incorrectly siphoned. | $8 \times 10^{17}$ | 37.5 hr | 3 people. Doses of 110, 43, 19 rads. | None reported. | None reported. |
| 07/24/64 | Scrap Recovery Plant, Wood River Junction, RI | Recovery of highly enriched uranium. | Solution hand-poured into unsafe geometry tanks. | $1.3 \times 10^{17}$ | 2 short pulses 1.5 hrs apart | 3 people. Doses of 10,000, 100, 60 rads. One fatality. | 20 percent of solution splashed out of tank. | None reported. |
| 08/24/70 | Windscale Works, England | Plutonium Processing | Plutonium accumulated in organic. | $3 \times 10^{15}$ | < 10 sec | 2 people. Doses of 2 and < 1 rad. | None reported. | None reported. |
| 10/17/78 | Chemical Processing Plant, INEL, ID | Solvent extraction column. | Loss of chemical control. | $3 \times 10^{18}$ | 15 min | Less than .13 rem. (In shielded cell.) | Air monitors detected considerable activity T 1/2 < 1 hr. | Filters removed most particles. |

Table 11.2 Previous Events at the GE Uranium Recycle Unit

| Date | System | Incident Description | Cause |
|---|---|---|---|
| 2/05/91 | Fluoride Waste | During the regeneration of the "B" IX column, about 33 kgs of uranium stripped from the column and transferred to tank V-106 (65,000-gal., unfavorable geometry). | Personnel Error: Valve in flow control instrumentation loop not reopened after calibration. |
| 9/18/90 | Radwaste | The failure of a crossover filter resulted in about 140 kgs of uranium being discharged to the site aeration basin via a continuously drained 5000-gal. unfavorable geometry tank (V-632). | Instrument Failure: (1) Plugged sampling system and low flow transmitter were inoperable. (2) The cross flow filter pressure detection system failed to detect the filter breakthrough. |
| 5/27/90 | Radwaste | Plugged overflow lines from two tanks caused unfiltered material to backup into vent-off gas line. | Design Error: Uncondensed steam flow to radwaste tank. |
| 5/17/90 | Radwaste | A small quantity of liquid collected in the daily composite sample. | Personnel Error: Valve in flow transmitter line left closed after calibration. |
| 4/13/90 | Radwaste | Composite sample measurements exceeded the nuclear safety engineering limit (86 ppm versus 25 ppm uranium). | Design and Sample System Instrument Failure: (1) Vent line (VOG) to the aging tanks was found plugged. (2) Sample line plugged and flow indicators were not operational. |
| 10/89-1/90 | Fluoride Waste | Carbonate filter breakthrough resulting in elevated uranium concentrations in tank V-106. | Instrument Failure: Sampling system uranium monitor was inoperable. |
| 1/12/90 | Nitrate Waste | Spill of 50-100 gal. of nitrate waste on the ground > 500 ppm $NO_3$ with a pH of 2.75. | Personnel Error: Wrong gasket installed in valve. |

Table 11.2 Previous Events at the GE Uranium Recycle Unit

| Date | System | Incident Description | Cause |
|---|---|---|---|
| 11/15/89 | Radwaste | Chemical upset resulting in about 64 kgs of uranium being discharged to the site process sewer lines via an unfavorable geometry tank. | Instrument Failure: Malfunctioning solenoid fill valve on fluoride quarantine tank and plugged radwaste sample line. |
| 10/17/89 | Nitrate and Radwaste | Both systems were experiencing cross flow filter problems and were in a rework mode for several hours which resulted in high tank levels and subsequent overflows to the roadway. | Unknown |
| 6/16-17/89 | Secondary Waste Treatment | Over addition of sulfuric acid into process lagoon (pH = 2.0 versus normal values of 6-9). | Personnel Error: Acid addition valve stuck open when the monitoring system was disconnected. |
| 9/1/87 | Radwaste | Uranium contamination found in non-controlled areas. | Plugged Sewer Line. |
| 6/30/86 | Radwaste | Higher-than-expected composite sample measurement (60 ppm uranium verses 3.0). | Equipment Failure and Personnel Error: Cross flow filter failure and plugged filter in sampling system (wrong size installed). |
| 5/19-20/86 | Solvent Extraction | Organic solvent found in the wash station sump. | Equipment Failure and Personnel Error: Failed check valve and block valve left open. |
| 5/7/86 | Solvent Extraction | Approximately 100 gallons of UNH solution at 195 g U/l was discharged to the UNH floor. | Personnel Error: Sample valve left open. |
| 3/9/85 | Solvent Extraction | About 450 gallons of aluminum nitrate solution drained to the aqueous waste Make-up Unit. | Equipment Failure: check valve internals missing. |

### 11.2.1 NRC Information Notice 90-63, "Management Attention to the Establishment and Maintenance of a Nuclear Criticality Safety Program"

IN 90-63 discussed an incident that resulted in a high concentration of highly enriched uranium being transferred to an unfavorable geometry waste collection tank. This IN was provided to alert licensees about an incident that resulted from inadequate management attention to the establishment and maintenance of a nuclear criticality safety program. IN 90-63 also mentioned that the licensee's inadequate evaluation of IN 89-24, "Nuclear Criticality Safety," dated March 6, 1989, may have been a contributing factor to the incident. IN 90-63 emphasized the need for continuing vigilance in providing a sound nuclear safety program. The industry was encouraged to review the following recommendations.

- Eliminate sumps and install piping to transfer waste solutions, thereby, eliminating the use of 11-liter cylinders in all process areas.

- Evaluate the procedures and practices for affixing labels to 11-liter cylinders in all process areas.

- Install in-line detectors and totalizers on all streams to waste collection tanks containing Raschig rings. Consider automatic shut off of the flow when detected uranium concentrations exceed an acceptable nuclear criticality control limit.

- Install additional controls on all streams to the collection tanks without Raschig rings. Evaluate interlock valves, as well as valves controlled by in-line detectors or conductivity meters connected to the alarm system.

- Develop material for, and train, first responders to unusual events.

- Retrain supervisory personnel on issues important to safety, labor relations, training, and emergency response.

- Evaluate the existing training program to ensure that personnel are trained and knowledgeable about assigned tasks in waste processing areas and nuclear criticality safety issues, including selected criticality accident histories.

- Reevaluate all nuclear criticality safety analysis to ensure proper application of the double contingency principle, with emphasis on unsafe geometry vessels.

- Reevaluate the audit and inspection programs to ensure that management control systems are being properly implemented.

- Review operating procedures for accuracy and completeness.

- Retrain personnel in complying with procedural requirements, with emphasis on mandatory compliance.

### 11.2.2 General Electric's Internal Evaluation of IN 90-63

GE's review and evaluation of IN 90-63 did not identify any unique failure modes which were not already addressed by their criticality safety program and determined that: (1) the plant did not authorize the type of uranium solution transfers describe in the IN, (2) controls employing Raschig rings were not used for criticality safety at GE., (3) an in-line uranium monitor was already functional on the fluoride waste discharges and the need for additional monitors was already being evaluated when IN 90-63 was issued, (4) an action plan and schedule to install additional uranium monitors on the Radwaste and Nitrate Waste streams had subsequently been developed to strengthen controls in these areas, in part due to the recommendations in IN 90-63, (5) the double contingency principle was believed to have been properly applied to unsafe geometry vessels, and (6) several activities had already been initiated to ensure better compliance with procedural requirements.

GE's internal evaluation of IN-90-63 exhibited a cursory and short-sighted effort. In fact, a number of items which applied to their facility and its operation were not adequately addressed, as evidenced by the May 29, 1991 incident:

- Although GE had double contingency on unfavorable geometry tanks, management did not assure that double centrifuging concepts were being implemented as intended.
- Operator training programs do not address or measure operator's knowledge level of assigned tasks and criticality safety issues.
- Audit and inspection programs do not address management control systems.
- Training programs for plant personnel with procedural requirements do not address or emphasize compliance.

## 11.3 Details of Precursor Events at GE

This section describes precursor events to the incident that occurred on May 29, 1991, at the Uranium Recycle Unit (URU) of the General Electric (GE) Nuclear Fuels and Component Manufacturing (NFCM) facility. During the incident, materials with uranium concentrations higher than safety limits were transferred to an unfavorable geometry tank. The solvent-extraction process continued to operate during attempted repairs of a failed level-control valve as did the transfer of solvent-extraction aqueous waste from quarantine tanks to an unfavorable geometry waste accumulation tank with (1) questionable sample analyses results and (2) no sample analyses performed. The following precursors were considered to be caused primarily by weaknesses in plant equipment design and performance, procedural compliance, adequacy of procedures, management oversight, and personnel performance.

### 11.3.1 Fluoride Waste System: 2/5/91

Approximately 33 kgs of uranium were stripped from a fluoride waste-ion exchange column and transferred to an unfavorable geometry tank, V-106. The incident was caused by personnel error. The technician closed a manual valve in a flow control instrumentation loop, as required, for loop calibration. However, the manual valve was not re-opened after calibration. Trapped pressure in the flow transmitter line down stream of the closed valve resulted in a false flow signal of 12 to 15 liters/minute being sent to a flow control loop. The flow demand set-point was 30 liters/minute. As a result, the flow control valve opened fully to try to reach the demand set-point value and approximately 5000 liters of ammonium bicarbonate (ABC) was flushed through the ion-exchange column to tank V-106. The normal process volume of ABC through the ion-exchange column is 375 liters. The 5000 liters of ammonium bicarbonate effectively stripped the ion-exchange column of uranium. The uranium in tank V-106 was reprocessed back through the ion-exchange system. The licensee's root cause determination and corrective actions did not include a comprehensive criticality safety review to evaluate the adequacy of the criticality safety controls.

### 11.3.2 Radwaste System: 9/18/90

The licensee estimated that no less than 30 kgs and no more than 300 kgs of uranium were discharged from the radwaste system through the plant sewage system to the aeration basin. The licensee determined that the cause of the incident was the failure of two criticality control functions in the radwaste system. The cross-flow filter pressure differential monitoring system had an alarm set-point that was greater than the change in pressure detected when the cross-flow filter failed. The residues that were discharged through the broken cross-flow filter clogged the lab sample line filter. In addition, the recently installed lab sample meters for line flow detection had not yet been calibrated. As a result, the loss of flow in the sample line was not evident in the lab. Interim lab sample line flow checks failed to detect the problem. The proportional or composite sampler on the discharge side of the radwaste system was not functioning properly.

The licensee's investigation report focused on process ramifications and recovery from the incident. There was no mention in the report of a re-evaluation of the criticality safety controls that obviously failed, or of their effectiveness. It appears that the radwaste favorable geometry and unfavorable geometry tanks were being operated in a valve-open pass-through mode prior to the incident rather than in a batch mode, as designed.

### 11.3.3 Radwaste System: 5/27/90

A two-foot section of plugged overflow line from two radwaste tanks caused a back flow of unfiltered, uranium-bearing material through a vent-off gas header into a safe geometry tank. Overflow line plugging was thought to have been caused by the admission of steam into the radwaste system. The licensee's root cause evaluation did not address possible problems with the radwaste tank's level indication system, design inadequacies with the overflow and vent lines, or the effectiveness of the criticality controls.

### 11.3.4 Radwaste System: 5/17/90

The accumulation of an unusually small quantity of liquid in the URU radwaste composite sampler bottle was caused by a closed inlet valve to the flow transmitter. Without an indicated flow from the flow transmitter, the solenoid valve that allows a small amount of liquid to flow into the sample collection bottle would not open. The analysis of samples taken from the radwaste permeate tank on the previous two days indicated no elevated uranium concentrations. The cause of the closed inlet valve was not determined and the possible generic implications with procedural and training inadequacies were not determined. The licensee did not determine the underlying cause of the closed inlet valve or if possible generic implications existed with procedural and training inadequacies as they relate to criticality control safety.

### 11.3.5 Radwaste System: 4/13/90

An analysis of a composite sample of the radwaste discharged into the plant sewage system revealed that the Nuclear Safety Engineering (NSE) uranium-concentration limit had been exceeded: 86 ppm of uranium versus the 25 ppm of uranium permitted. It was determined that the overflow lines on the aging tanks were plugged, causing unfiltered sludge to backup into the tank vent lines. The sludge flowed down the vent lines into a radwaste permeate tank and was then transferred into the permeate receiver tank. The sludge was subsequently transferred into the permeate holdup tank and finally into the plant sewage system. The amount of uranium-bearing materials that were discharged into the plant sewage system was analyzed and found to have exceeded allowable limits. The automatic sampling system did not detect the high uranium content in the permeate receiver tank because the sample line was found to be partially plugged, in addition to the nonfunctioning flow indicators. It was assumed that the sample/analyzer system was analyzing liquids trapped in the line downstream of the plugged portion of the line. The adequacy of the sample system design as it affects the criticality controls was not reviewed as part of the licensee's root cause determination.

### 11.3.6 Fluoride Waste System: 10/89 - 1/18/90

On January 18, 1990, the licensee discovered that twice between October and December 1989, uranium concentrations in excess of 200 ppm were found in the unfavorable geometry fluoride waste collection tank V-106. The cause was believed to be the failure of a filter, which allowed materials with higher-than-normal uranium concentrations to be transferred to quarantine tanks. The quarantined materials with high-uranium content were transferred to the V-106 tank because the fluoride liquid waste uranium monitor in the fluoride waste quarantine system was inoperable at the time the filter failed. It was later determined that a new uranium monitoring system was still in the testing stage at the time of the incident. In addition, the failed filter had been installed without a facility change request (FCR) and Nuclear Safety Engineering (NSE) review and approval. When the FCR was written and submitted to NSE (1/24/90), it was incorrectly identified as a Nitrate Waste Monitor rather than or a Fluoride Waste Monitor. The Nuclear Safety Release/Requirements procedure (NSR/R) regarding the maximum allowable uranium concentration limits that could be transferred to the V-106 tank from the quarantine tank had two values: 150 ppm for operators and 300 ppm that could be used with supervisor approval. In addition, the NSR/R did not have an allowable limit for uranium concentrations in tank V-106. These two items caused confusion as to whether uranium concentration limits had been exceeded.

The design and operation of the fluoride waste system is such that high uranium concentrations in tank V-106 caused the ion-exchange process to cycle more frequently. The regeneration portion of the ion-exchange process resulted in additional materials containing uranium and ammonium bicarbonate being deposited in tank V-106. The fluoride waste system's regeneration/recovery operations caused additional high uranium concentrations to be deposited in the quarantine tanks, where the cycle started again. These inter-system links had not been considered in the criticality safety analysis.

### 11.3.7 Nitrate Waste System: 1/12/90

A block valve gasket failure resulted in a spill of approximately 50 to 100 gallons of nitrate waste water with a pH of 2.75 and $NO_3$ content > 500 ppm outside the URU facility buildings. The licensee determined that the amounts of $NO_3$ and the pH levels were within specified limits and that, therefore, the failure had no impact on the environment. An improper gasket installed on the valve was determined to be the root cause of the gasket failure and resultant spill. However, the licensee did not determine the reasons why an improper gasket was installed and whether there were any generic implications associated with the incident.

### 11.3.8 Radwaste System: 11/15/89

An analysis of a composite sample of radwaste discharges revealed a uranium concentration of 866 ppm. Based on the volume discharged, the licensee determined that approximately 64 kgs of uranium were discharged into the site process sewer lines through unfavorable geometry tank V-632. The high uranium content was believed to have been caused by the intrusion of carbonates from fluoride waste system tanks into the radwaste system. A malfunctioning solenoid valve which controlled a fill valve to a fluoride quarantine tank caused the fill valve to remain partially open at all times. This allowed the quarantine tank to overfill and drain into a sump which discharges to the radwaste system. The overfill incident occurred during a period of high carbonate concentrations from an ammonia recovery process. It was assumed that the carbonate caused uranium to precipitate from the radwaste aging and permeate tanks. The permeate tanks discharge to the process sewer lines. The high uranium concentrations in the radwaste system were not detected by the automatic sampling system because the sample line was partially plugged. The licensee performed a limited review of the adequacy of the design of the sampling system and did not evaluate the criticality control system.

### 11.3.9 Nitrate Waste and Radwaste Systems: 10/17/89

Nitrate waste and radwaste treatment tanks overflowed into the roadway outside the URU facility. The nitrate waste and radwaste systems were placed in a recycle mode for several hours because of cross-flow filter problems. Recycling resulted in high tank levels and overflows. Lab analysis results of samples taken from the spill indicated a concentration of 32 ppm of uranium. The licensee did not evaluate the nitrate waste and radwaste tanks level indication and criticality safety control systems.

### 11.3.10 Sulfuric Acid Addition and Process Lagoon Systems: 5/16-17/89

Samples from the process lagoon were found to have very low pH values (2.0 actual versus 6-9 normal pH values). The low pH values were determined to have been caused by the addition of an excessive amount of sulfuric acid to the lagoon because of an inoperable autocall (pH monitor/alarm) system in the sulfuric acid addition system. Analysis of samples taken from the Cape Fear River indicated normal pH values. The autocall system was found disconnected at the lagoon. The licensee determined that the system had been disconnected prior to lagoon dredging operations and was not reconnected after dredging was completed. Subsequently, maintenance personnel discovered the disconnected autocall system and reconnected it but failed to notify the licensee's operations or regulatory compliance personnel of their findings and actions. (An incident of this type (GE-Class II) requires that the NRC be notified within 10 days. GE notified NRC within these time limitations.)

### 11.3.11 Radwaste System: 9/1/87

A backup overflow from a radwaste process discharge line resulted in the contamination of a non-controlled area of the Aqueous Solution Make-up Unit (ASMU). The cause of the overflow was determined to be a partially clogged process discharge line that prevented a large volume of material from being discharged from the URU in a relatively short period of time. The uranium concentration of the material found in the bottom of the process line was measured to be 2680 ppm (allowed limits were 25 ppm of uranium). The clogged line was thought to have been caused by the gradual build up of precipitated uranium-bearing lime solution that was not filtered out from the radwaste processing system. During clean-up activities, the levels of contamination spread in some instances because uranium trapped in the lime leached out when the acidic cleaning material came into contact with the dried spillage. The licensee did not determine the quantity of uranium deposited in the process lines and the ASMU area. Corrective actions included routine inspections of process sewer lines to identify any plugging. No evaluation of the controls to prevent recurrence was evident nor was the criticality safety analysis reviewed to determine if the controls were still adequate.

### 11.3.12 Radwaste System: 6/30/86

Higher than expected uranium concentrations (60 ppm actual versus 3 ppm expected) were found in a radwaste composite sample and in the permeate receiver tank. A cross-flow filter failure and a blockage of the sampling system filter were the primary causes of the incident. A 5-micron filter rather than the 50-micron filter specified was installed. Because the automatic sampling system did not detect the high uranium concentrations, the radwaste process continued to ingest materials with high uranium concentrations from the failed cross flow filter. (Nuclear Safety Engineering (NSE) was concerned that there was no way to verify the sampling system filter size/type once it had been removed from its packaging. NSE was also concerned that the cross-flow filters performance was not adequately monitored. The accumulation of post-filter material in the receiver and holdup tanks was a concern because the system's safety analysis was based on the absence of precipitation material in the tanks.)

In response to NSE's concerns, the process engineer indicated that the filter problem would be resolved by instructing operators on the installation of proper filters and by stocking the correct filters. The process engineer indicated that turbidity meters had been used but were not reliable and were difficult to calibrate and maintain. Also, the filter flux calculation used as the active engineering control would be changed to increase the sensitivity without reaching the point of getting false indications and compromising the system's credibility. The turbidity meter issues and filter flux calculation sensitivity adjustments were turned in to engineering for further study.

### 11.3.13 Solvent-Extraction System: 5/20/86

Organic solvent used in the solvent-extraction process was discovered in a URU wash station sump. The concentration was determined to be 100 g U/liter. The solvent reached the sump because of a failed check valve and an open block valve in an open organic waste line in the URU lab which was connected to a solvent-extraction surge tank. The block valve had not been closed after a small centrifical pump was removed from the sample system. The pump-to-line connection was not capped. The failed check valve and the open block valve and line allowed organic solvent from the solvent-extraction process surge tank to back flow through the line and into the lab drainage system. The lab drainage system ultimately drains into the nitrate waste at the wash station sump. Samples from the nitrate waste permeate and the nitrate waste V-103 tank were checked for <u>visible</u> signs of solvent and none were noted. Twenty buckets of organic and aqueous solution were pumped from the wash station sump and 15 gallons of solvent were recovered. Lab personnel were instructed to be more aware of and cautious about equipment conditions and system line-ups in their areas of responsibility. However, there was no apparent follow-up as to why the valve in the solvent pump discharge line was left open nor was a review of the criticality safety analysis performed to determine if controls were adequate.

### 11.3.14 Solvent-Extraction System: 5/7/86

Approximately 100 gallons of UNH solution with a concentration of 195 g U/liter were discharged onto the UNH storage area floor. Approximately two to three gallons seeped into the URU manager's office and about three gallons reached the floor of the $UF_6$ cylinder truck bay. The cause of the incident was principally due to open in-line manual valves and an open drain valve in a transfer line on a rework tank. The drain valve was located on the suction side of the transfer line's pump and had a flexible hose attached to it. As the UNH solution level in the tank reached the transfer line elevation, it carried over through the transfer line, the open valves, and out the flexible hose and onto a wall, where it splashed down onto the UNH storage area floor. Normally, a control room operator (CRO) would have inspected the tank and line alignment before the transfer to assure that the tank and attached piping/valve alignment was correct. However, the single CRO on duty was unable to leave the control room. Instead, a shift engineer inspected the system arrangement and decided that it was acceptable. Corrective actions included the removal of flexible hoses, installation of a check valve in the transfer pump's discharge line, the application of sealant to the UNH storage walls, assurances that operations personnel perform all operations on the floor and that adequate coverage is provided in the control room. The licensee did not perform a comprehensive criticality safety review to evaluate the adequacy of the criticality safety controls.

### 11.3.15 Solvent-Extraction System: 3/9/85

Approximately 450 gallons of aluminum nitrate solution drained to the floor in the north room of the Aqueous Solution Make-up Unit (ASMU) area from make-up tank T-700. (Before the incident, an aluminum hydroxide/nitric acid solution was being heated to 100° C by the direct injection of steam.) The area was cleared of fumes and washed down with water. The cause of the incident was attributed to nitric acid eroding carbon steel piping upstream of a stainless steel check valve. Subsequent examination of the check valve revealed that the internals were missing. The licensee determined that when the steam control valve was closed, nitric acid back-flowed into the steam line and through the check valve body. The licensee did not determine the reasons why the check valve internals were missing.

## 11.4 Solvent-Extraction System: Prior to Incident

Because of the lack of management oversight and the lack of attention to detail, the following indicators provided missed opportunities for the licensee to identify plant problems that may have prevented this incident.

- On May 27, 1991, at about 10:10 a.m., the contents of quarantine tank (QT) V-290 were transferred to tank V-103 without a sample measurement.

- On May 28, 1991, at 12:25 p.m. and 2:30 p.m., the contents of QT tank V-290 were transferred to tank V-103 without a sample measurement.

- With prior knowledge of sample loop problems that have occurred in other URU systems, the licensee continued to operate the aqueous waste (AW) sampling loop without an operational flow-sensing device.

- Modifications made in 1987 and 1990 included the addition of the head-end concentrator (HEC) condensates and product reboiler condensate to the AW quarantine tanks V-290 and V-291, respectively. These condensates added about 345 additional gallons per hour (gph) to the quarantine system. The AW flow rate to the QTs was a nominal 50 gal/hr, and the volume of quarantine tanks V-290 and V-291 were 600 gallons each. This modification had the effect of making it difficult if not impossible for the operator to follow the proper sampling and analysis procedure and keep the solvent-extraction system running during upset conditions.

- The density monitors had been removed from waste accumulation tank V-103 and there were no requirements for density sampling before its contents were transferred to tank V-104 at the Waste Treatment Facility (WTF).

- The licensee had not established a program for reviewing data from control room operator, chemistry lab, and computer output logs to determine if the aqueous waste system was being operated in accordance with established procedures and nuclear criticality safety requirements. This deficit was a reflection of: (1) the licensee staff's general attitude that a criticality accident could not occur at the facility since only low enriched uranium was being processed there and (2) management's very limited oversight of daily in-plant operations.

## 11.5 Findings

The team's evaluation of these precursor events indicates that the licensee's program for determining the causes of incidents and the corrective actions to prevent their reoccurrence was inadequate. Repeated failures with plant process equipment, the subsequent failures of the sampling system equipment used to detect the process problems and, in some instances, the personnel errors of commission and omission causing equipment and system failures, have all contributed to significant quantities of uranium-bearing materials being deposited in unfavorable geometries. The licensee's root cause determinations and corrective actions generally did not include comprehensive reviews to evaluate the adequacy of facility criticality safety controls.

# 12 REGULATORY ASPECTS

The General Electric (GE) Nuclear Fuels and Component Manufacturing Operation near Wilmington, N.C., is licensed by the U.S. Nuclear Regulatory Commission (NRC) pursuant to the regulations specified in Title 10 of the *Code of Federal Regulations*, Part 70 - Domestic Licensing of Special Nuclear Material. The company holds license number SNM-1097, expiration date June 30, 1989. The licensee submitted an application for license renewal and is currently operating under applicable renewal provisions.

## 12.1 General Observations on License and Licensing Process

### 12.1.1 Licensing History

Initial operations with enriched uranium at the Wilmington site were authorized by Atomic Energy Commission (AEC) License SNM-1097, dated January 13, 1969. The NRC renewed the license on May 24, 1976, for a five-year period. On April 28, 1981, GE (the licensee) filed an initial application for renewal of license SNM-1097, and on May 27, 1981, the licensee submitted a revised application for renewal. On February 1, 1983, the NRC staff asked for extensive revisions of the renewal application and, as a result, the NRC extended the expiration date to January 31, 1984. Following a review of the revised renewal application and several supplements, the NRC renewed the license on June 29, 1984, with an expiration date of June 30, 1989. On May 22, 1989, the licensee submitted a renewal application. This application was under review by the NRC as of the date of this report.

### 12.1.2 Current License

Since its issuance, the current license has been amended 19 times. A summary of these amendments is shown in Table 12.1. Activities authorized by the license include: $UF_6$ conversion, fuel manufacture, scrap recovery, fuel process technology operations, laboratory operations, general service operations, waste treatment and disposal, and uranium recovery from lagoon sludge. Other significant conditions of the license include authorization to possess a limit of 50,000 kgs of uranium compounds at any enrichment below 6 percent of uranium-235. The license also includes a possession limit of 500 kgs of uranium, in any form, enriched from 6 percent to less than 10 percent for use in developmental process technology operations.

### 12.1.3 The NRC Licensing Process

Fuel fabrication facilities are licensed by NRC's Office of Nuclear Material Safety and Safeguards (NMSS). NMSS licensing responsibilities include reviews of new licenses, license amendments, and license renewals. NMSS conducts a formal safety evaluation for each of these licensing actions. In this evaluation process, NMSS provides the NRC regional office, which has inspection responsibility for the facility, an opportunity to review and comment on the licensee's proposal. Upon determination that the license application satisfies applicable requirements of 10 CFR 70, and when there is a high degree of confidence that operations can be conducted safely, NMSS approves the licensing action. In the past, licenses have been issued for a five-year period. Under current licensing policy, some licenses are being issued for a ten-year term.

Fuel facility licensing as currently administered requires that the licensee submit a two-part license application. Part I contains the licensee's proposal for those items that would be License Conditions and would be legally binding on the licensee. These License Conditions are somewhat analogous to Technical Specification (TS) commitments for power and research reactors, but generally lack the specific details found in TSs. Part II of the application contains the Safety Demonstration, which is somewhat analogous to the Final Safety Analysis Report and technical bases contained within TSs for power reactors. The Safety Demonstration, although often containing significant technical information on operations and controls, is (1) not incorporated as part of the license or license conditions, (2) not legally binding on the licensee, and (3) not enforceable under the provisions of the current NRC Enforcement Policy.

## 12.2 Regulatory Oversight Problems

The team evaluated the adequacy of requirements and the adequacy of NRC's review of requirements as it relates to preventing or responding to a potential criticality event. Based on this evaluation, weaknesses were identified with respect to regulations and regulatory guidance, licensing, and the inspection program. The team found that these weaknesses in regulatory oversight had the effect of contributing to the gradual erosion of safety margins at the licensee's facility.

### 12.2.1 Emergency Planning

A reading of the licensee's radiological contingency emergency plan and its implementing procedures does not clearly lead to an emergency classification and response actions for a potential nuclear criticality. The licensee plan and procedures do give clear guidance concerning classification and response actions for an actual criticality. The planning standards in the regulations, 10 CFR 70.22, and the regulatory guidance in Regulatory Guide 3.42, do not address planning considerations for potential nuclear criticalities.

NRC's review of the licensee's emergency plan did not reveal this problem. Guidance available to NRC reviewers is also deficient with respect to emergency planning considerations of potential nuclear criticalities.

### 12.2.2 Licensing of the Uranium Recycle Unit (URU)

The licensee's submittal, dated December 3, 1984, for the use of the URU facility was authorized for operation under License Amendment No. 3, dated February 26, 1985. The licensee's submittal was incorporated by reference in its entirety into License Condition No. 9. The NRC licensing staff intended that the commitments, statements, and limits regarding criticality controls as contained in the submittal could only be changed by a license amendment. In contrast, the licensee did not consider the incorporation of the December 3, 1984, submittal into License Condition No. 9 as limiting their ability to make changes in the process controls as normally done under the administrative provisions of their license. Such previous documents have been considered as Part II, "Safety Demonstration," of the license of which the licensee is authorized to make changes without NRC approval. NRC inspectors also had the same understanding as the licensee and did not challenge the licensee's unilateral authority to make criticality control modifications in the URU. As noted in Section 12.2.4, the inspectors focus was primarily on assuring that such modifications were accomplished through the licensee's approved Facility Change Request process.

Incorporation of a licensee's submittal in support of a major modification amendment request similar to the licensee's December 3, 1984, document as a license commitment in its entirety, is highly unusual for a broad scope licensee. A broad scope licensee, as the phrase implies, has the authority to make safety control modifications. The limits of this authority are defined in the Part I of the license, as it is in Section 2 of Part I in the licensee's license. The team concludes that the limit of the licensee's authority to make changes in URU criticality controls was not mutually understood among the licensee and the NRC licensing and inspection staffs.

The NRC licensing staff did not convey to the licensee or to the inspection staff the intent that all statements and representations contained in the December 3, 1984, submittal cannot be modified except by license amendment.

### 12.2.3 Precursor Incidents

The team noted that the NRC did not have a well-defined review program for fuel facility incidents and the reporting threshold for licensee incidents is too high. Even if the reporting threshold were the same as that used by the licensee to initiate their investigation process, it may have prompted a more thorough review by the NRC. This thorough review might have recognized the recurrence of criticality control failures and prompted regulatory action that would have precluded this incident.

Because there were no formal reporting requirements, the review of licensee incidents was left up to the discretion of the NRC inspector. The precursor incidents were reviewed during routine inspections, but not in a manner that would reveal trends. Inspection guidance is relatively silent on the valuable insights of operations that such incident reviews would reveal. In addition, there is no specific inspector training in this regard.

### 12.2.4 Facility Change Requests (FCR)

Under the provisions of the license, the licensee is authorized to make changes to systems, processes, and facilities via a Facility Change Request system. The licensee has developed internal procedures to provide guidance on the initiating, reviewing, and implementing of a change request. A weakness was noted in the licensee's procedures that implement the FCR process. Initiation of the FCR process has a high threshold, and is not always used when changes are made. The decision as to whether a change rose to the level of needing an FCR and, thus, a nuclear safety review, was made by the staff member making the change. There were no audits to assure proper decision making at this level within the organization.

The regulations in 10 CFR 70 do not address facility changes and changes of procedures and methods; i.e., there is no regulation comparable to that specified in 10 CFR 50.59, "Changes, tests, and experiments." Although the regulations in Part 70 do not explicitly address change reviews, they are handled on a case-by-case basis during the development of license conditions. The current license (Section 2.7 of Part I) addresses the change review process and provides criteria for making such changes. The criteria in license conditions are too broadly stated. Consequently, as stated above, this promoted the exercise of discretion on the part of the licensee in establishing the need for change reviews.

NRC inspection procedures provide limited guidance to inspectors for the review of facility changes. The team determined that NRC inspections tended to focus on the licensee's administration of the FCR process and not on the quality of the nuclear criticality safety analyses that support the FCR. In addition, NRC inspections (and inspection guidance) do not address actions to uncover and rectify improper decisions concerning use of the FCR process by the licensee.

### 12.2.5 Procedural Compliance

The team reviewed NRC's nuclear criticality inspection procedures and selected NRC nuclear criticality safety inspection reports. It was noted that the inspection procedures, which offer guidance to inspectors, were lacking with respect to details concerning verification of the licensee's implementation of operations procedures. Further, it was noted that NRC inspection reports did not appear to address inspection of the licensee's implementation of operating procedures. Procedural compliance seemed to be a matter which was left to the discretion of the licensee's internal audits to review. The team verified through interviews with the NRC inspection staff that operating procedure compliance is not covered in the routine inspection program.

### 12.2.6 Management Oversight

The team found that the NRC inspection program did not focus in depth on licensee management oversight of licensed activities. The NRC inspections tended to verify that the licensee has controls (e.g., organization structure, procedures, audits), but did not focus on the effectiveness of these controls. For example, the quality and substance of issues addressed in licensee audits are not examined in detail. Inspection Procedure 88005 addresses the types of management controls to be reviewed. However, it tends to stress the verification of controls, as opposed to the effectiveness of controls. The knowledge and skills of individual inspectors, along with NRC management direction, also has influence on the degree of evaluation of licensee management controls.

The team further noted that the current license conditions relative to management controls are rather broadly written. Examples include nuclear criticality safety audits and nuclear criticality safety training.

## 12.3 Findings and Conclusions

**Regulations and Guidance**

- Emergency planning development does not address potential criticality incidents as a specific emergency situation to be considered.

- The NRC incident reporting criteria threshold is too high. A structured, formal NRC review of the licensee's incident investigation reports may have revealed the ineffective criticality controls.

**Licensing**

- The limit of the licensee's authority to make changes in URU criticality controls was not mutually understood by NRC and the licensee.

- Licensee conditions, in general, are too broadly written, particularly in the management oversight area.

**Inspection Guidance and Implementation**

- The criticality safety inspection guidance and implementation did not focus on operations procedural compliance.

- The inspection program focused on administration of the Facility Change Request process and not the quality of the Nuclear Safety Analysis that supported approving the Facility Change Request.

- Inspections did not focus on ensuring that licensee management maintain appropriate oversight of licensed activities.

Table 12.1 Amendments to License SNM-1097

| Amendment | Date | Description |
|---|---|---|
| 1 | 12/11/84 | Revised Radiological Contingency and Emergency Plan. |
| 2 | 12/17/84 | Authorized use of UNH conversion process. |
| 3 | 02/26/85 | Authorized use of UPMP processes. Condition 12 added to license, amendment #2 superseded. |
| 4 | 03/04/85 | Authorized use of the Calcium Fluoride Waste Treatment Facility. |
| 5 | 07/12/85 | Revised subsection 1.8.4.1. |
| 6 | 08/09/85 | Authorized test programs for recovery of uranium from nitrate lagoon and revised Radiological Contingency and Emergency Plan. |
| 7 | 11/07/85 | Authorized the disposal of industrial wastes products containing uranium. |
| 8 | 05/27/86 | Name change, abandonment of warehouse, and revised Radiological Contingency plan. |
| 9 | 09/19/86 | Changed Radiological Contingency Plan. |
| 10 | 06/11/87 | Granted exemption to 10 CFR 20.203(f) on container labeling. |
| 11 | 07/23/87 | Prevention of hydrocarbons from entering $UF_6$ cylinders. |
| 12 | 11/12/87 | Amended Part 1 license conditions. |
| 13 | 12/14/87 | Changed Radiological Contingency and Emergency Plan. |
| 14 | 05/12/88 | Authorized transfer of test quantities of $CaF_2$ to vendors. |

Table 12.1 Amendments to License SNM-1097

| Amendment | Date | Description |
|---|---|---|
| 15 | 06/27/88 | Dry $UF_6$ Conversion. |
| 16 | 03/10/89 | Revised possession limits for uranium to permit GE to remain a MC&A Category III facility. |
| 17 | 03/20/89 | Revised Radiological Contingency Plan; added facility manager. |
| 18 | 01/12/90 | Authorized transfer $CaF_2$ to briquette manufacturer. |
| 19 | 01/11/91 | Authorized utilization of Uranium Recovery Lagoon Sludge System. |

# 13 FINDINGS AND CONCLUSIONS

## 13.1 GE-Wilmington Fuel Manufacturing Operations

The incident at the General Electric (GE) Nuclear Fuel and Component Manufacturing (NFCM) facility began during the second (3:00 p.m. to 11:00 p.m.) shift at approximately 8:30 p.m. on May 28, 1991, by an upset in the solvent-extraction (SX) process in the Uranium Recycle Unit (URU). A malfunctioning level-control valve for the SX A Column interface caused uranium-bearing feed material to be carried into the waste treatment system. While attempting to keep the process running over a nine-hour period with the faulty level-control valve, licensee personnel transferred the contents of a number of waste tanks containing approximately 150 kgs of uranium enriched to 3.2 percent to the large volume, unfavorable geometry tanks V-103 and V-104. The control measures that should have precluded this incident from occurring failed. Once the licensee determined that 150 kgs of uranium had been transferred to tank V-104, Waste Treatment Facility operators ensured continued sparging operations to prevent the material from possibly settling into a nuclear critical configuration. The uranium was ultimately removed by centrifuging operations over a 4 1/2 day period. That the incident did not result in a nuclear criticality accident was attributable to the fact that the last remaining safety barrier, sparging, was successfully applied, rather than because the accident was precluded by indepth barriers and management control systems.

The team determined that three interrelated root causes contributed to the incident.

1. There was a pervasive licensee attitude that a nuclear criticality was not a credible accident scenario. While the licensee understood and recognized that a nuclear criticality with low-enriched uranium was technically possible, and that there were regulatory requirements to establish measures to guard against such an accident, the licensee's perception was that the risk was so low that a criticality accident inherently would not happen.

2. Licensee management did not provide effective guidance and oversight of licensed activities to assure that operations were conducted in a safe manner.

3. There was a deep-seated production-minded orientation within the licensee organization that was not sufficiently tempered by a "safety first" attitude, particularly regarding nuclear criticality safety.

The following specific findings made in the course of this investigation support the conclusions listed above. Although many examples are applicable to more than one conclusion, the examples are grouped under the causes for which they appear to be the most appropriate.

Pervasive Attitude that a Nuclear Criticality Accident Is Not Credible

- When the 150 kgs of uranium were determined to be in tank V-104, the licensee did not consider the incident significant enough to implement its Radiological Contingency and Emergency Plan (RCEP). Eight hours after becoming aware of the incident, the licensee made a "courtesy" call to the NRC. Fifteen hours later the NRC convinced the licensee that the matter was serious enough to activate the RCEP and declare an emergency Alert. Throughout the uranium recovery stage, the NRC had to vigorously urge the licensee to evaluate and plan for alternate contingencies if the sparging efforts were ineffective. Even by the time the Incident Investigation Team (IIT) arrived on June 2, 1991, licensee management still felt that the incident did not warrant such escalated attention. Further, the licensee's press release characterized the incident as a "chemical emergency."

- Both new employee and annual refresher training about general criticality safety, while meeting minimum regulatory requirements, stressed that no criticality accident ever occurred at a low-enriched uranium facility.

  Supervisors and senior operators provided on-the-job criticality control training for junior operators. Supervisors and senior operators themselves received no specific training to guide them in this particular task. Nor were audits conducted to determine the effectiveness of this job-specific criticality control training.

- Nuclear safety audits, while meeting minimum regulatory requirements, focused on changes to equipment and hardware but not on implementation of process controls for criticality safety. There were no audits to assure that procedures for criticality control sampling and analysis were followed and no records to support such audits were required either to be made or maintained. The adequacy and accuracy of criticality control laboratory analyses and process computer control system changes were not independently verified.

- Automatic criticality safety sampling and analysis systems which proved difficult to maintain were eliminated, as were interlocks and alarms, from the automatic criticality control system without an adequate review of the impact of their removal to criticality safety. Configuration control of the automatic control system was inadequate: No independent verifications were made to ensure that criticality safety considerations had not been compromised when changes to the system were made, nor was a distinction made between safety and process alarms, while some automatic shutdown functions were disabled.

Ineffective Management Oversight

- Over a year ago, the licensee reorganized the NFCM and eliminated many supervisory positions. The resulting reorganization left each supervisor with a wide span of control. In particular, during non-day shifts and weekends, the URU was one of five areas of responsibility for an Area Coordinator. If circumstances permitted, the Area Coordinator might get to the URU once per shift. Thus the Area Coordinator could not, and was not expected to, provide close supervision. In addition, there was no technical support on-site during non-daytime and weekend shifts.

- There were no verifications of the adequacy of operator performance relative to operating procedures. In addition, the licensee did not require that operators keep logs to facilitate operations audits. As a result, the logs that were kept were inconsistent and only noted what each operator felt was necessary to note at the time of an entry.

  Records of the logging system for the operating process control computer were not routinely reviewed and not retained for historical purposes. For example, computer logging records prior to May 18, 1991, were already discarded by the licensee at the initiation of this investigation.

  The operators were aware that there was little management oversight of operations and that they would come under scrutiny only if production did not meet management expectations.

- Design and process modifications that directed additional waste streams into the SX quarantine tanks made it very difficult for operators to comply with the sampling and analysis portion of their operating procedures while maintaining continuous operations. The operators found that by overriding automatic interlock systems and deviating from procedures they could operate within the constraints of rapidly filling quarantine tanks. This practice, in effect, ensured that the sampling and analysis of the quarantine tanks before release would be non-representative. Because of insufficient oversight, management was not aware of this practice.

- There was a lack of "ownership" for the URU and assurance that all activities were conducted in a safe manner. For example, individual staff members in the separate operation, engineering, maintenance, chemistry, instrumentation and calibration, and regulatory compliance components of the URU all reported to different peer organizations. These peer organizations all reported directly to the Manager, Nuclear Fuel and Component Manufacturing (NFCM). The Manager of NFCM has 12 direct and 4 indirect organizational managers reporting to him. As a result of this large span of control, his cognizance of specific URU activities is inherently limited.

- Equipment repairs and process modifications with nuclear safety implications were formally implemented by the Facility Change Request (FCR) process. The decision as to whether a change rose to the level of needing an FCR and, thus, a nuclear safety review, was made by the staff member initiating the change. There were no audits to assure proper decision making at this level within the organization.

When FCRs were processed, the nuclear safety reviews were generally limited to the change itself, without sufficient consideration being given to related systems or equipment and their criticality controls.

- The criticality controls established at the initiation of URU operations in 1985 were gradually eroded until the only viable barrier against a nuclear criticality accident in an unfavorable geometry waste tank was the sampling and analysis of the quarantine tank contents prior to release to the large volume tanks.

  Licensee management and the nuclear safety organization were apparently unaware of the adverse cumulative effect of the criticality control modifications and did not ensure the integrity of the quarantine tank sampling system. The team found no indication of pre-operational or periodic testing of the sampling system to ensure representative sampling. In addition, previous incidents at this facility were caused by inoperable or clogged sampling systems similar to the system used in the solvent-extraction system.

- Management oversight of maintenance activities was inadequate. There was no supervision except on the day shift during week days. An extremely liberal work control system was evident by the lack of governing work order procedures, informal work authorizations, lack of documentation for work performed, and the failure to consider the impact maintenance activities had on criticality controls. Engineering support of maintenance was also minimal, with no root cause determinations, common mode failure analyses, or trending performed on failed equipment.

- The size, qualifications, and experience of the Nuclear Safety Engineering staff is impressive. This resource has not been utilized adequately to ensure that criticality controls have been effectively implemented and maintained at the URU.

Production Versus a Safety-First Attitude

- The operators' overriding concern was to keep the process going, even if it meant not conforming to procedures or disregarding criticality controls.

- SX operating procedures emphasized troubleshooting activities for process upset conditions. Operators were creative in responding to this guidance. For example, when the rework tank volume was not available, a quarantine tank having contents with a higher uranium concentration than that allowable would be dumped to the process floor and returned to the system through a sump. To meet production demands, operators also sampled quarantine tanks while they were filling rather that waiting until they reached the sampling level prescribed in the procedures so that they could transfer the contents immediately when the tanks filled.

- Safety interlocks, shutdown functions, and alarms were eliminated from the automatic process control system because they were difficult to maintain and a bother to operations.

Approvals to authorize such changes were short-sighted, not adequately considering the impact on criticality safety.

- Licensee investigations of internal incidents focused on their process and recovery implications and not on failed or ineffective criticality controls, notwithstanding that the nuclear safety organization was always represented on the licensee's investigation teams.

- Maintenance priorities and resources focused on meeting continued plant operation goals. For example, almost 25 percent of the maintenance work orders completed were classified as Emergency by control room operators. The team determined that, at times, these emergency work orders were inappropriately applied to non-emergency equipment malfunctions that affected system performance or threatened its shutdown.

## 13.2 Regulatory Oversight

The team identified weaknesses in NRC regulatory guidance, licensing, and inspection programs that had the effect of contributing to this incident.

- The NRC regulatory guidance and review process for the development of a licensee emergency plan does not adequately address potential nuclear criticality scenarios. The licensee's RCEP and implementing procedures do not clearly lead to an unambiguous determination of the seriousness of such a situation.

- There was not a mutual understanding among the NRC licensing and inspection staff and the licensee about the licensee's criticality control commitments related to the URU.

- NRC's reviews of precursor incidents were not sufficient to recognize that a loss of control of large quantities of uranium was serious enough (1) to warrant notification of the NRC, and (2) to question the effectiveness of the facility's criticality controls.

- NRC inspection guidance and implementation focused on the administrative implementation of the licensee's Facility Change Request (FCR) process rather than on the quality of nuclear safety analyses supporting the FCRs.

- NRC inspection guidance and implementation did not focus on procedural compliance regarding the implementation of operational criticality controls.

- NRC inspection guidance and implementation did not focus on ensuring that licensee management maintained appropriate oversight of licensed activities.

# APPENDIX A

# INCIDENT INVESTIGATION TEAM CHARTER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 31 1991

MEMORANDUM FOR: Chairman Carr
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: INVESTIGATION OF MAY 29, 1991 POTENTIAL CRITICALITY SAFETY PROBLEM AT GENERAL ELECTRIC NUCLEAR FUEL PLANT INVOLVING EXCESSIVE URANIUM IN A WASTE TREATMENT TANK

On May 29, 1991 at 3:40 p.m. EDT, representatives of the General Electric Fuel Fabrication Plant in Wilmington, North Carolina, notified the NRC's Region II office of a potential criticality safety problem at the facility. The licensee identified a higher than allowable concentration of uranium in a process tank of the waste treatment system. Subsequently, on May 30, 1991 at 0638 EDT, the licensee declared an alert. The solvent extraction system, which apparently caused the situation, was shutdown and actions were initiated to reduce the uranium concentration in the tank. A Site Team, led by a Region II Branch Chief, supported by two Region II specialists was dispatched to the site and arrived at approximately 1:00 a.m. on May 30. That team was supplemented with additional specialists from NMSS and RII and a RII Public Affairs Officer to provide a complement of eight people as of May 31. The NRC entered the standby mode at 5:44 pm EDT on May 29, 1991 to support the Region II Incident Response Center in order to monitor the plant's recovery effort and support the team of specialists sent to the site. Because of the safety significance and the potential regulatory questions the event raises, I have requested AEOD to take the necessary actions to upgrade the current Site Team to an eight member NRC Incident Investigation Team (IIT) as soon as practical after the emergency condition is resolved at the site.

The IIT is to: (a) fact find as to what happened; (b) identify the probable causes as to why it happened; (c) make appropriate findings and conclusions which would form the basis for any necessary follow-on actions.

The team will report directly to me and is comprised of: Ross Scarano, (RV), Team Leader; William Cline, (RII), Deputy Team Leader; Richard Correia, (NRR); Charles Hooker, (RV); Robert Wilson, (NMSS); Hulbert Li, (NRR); Sonia Burgess (RIII); and Edwin Fox (NRR). Because of the limited number of technical experts able to investigate an event of this type, some team members have had previous inspection experience at the facility. Enclosed is the charter for the IIT to use in the review of the event.

The IIT was selected on the bases of their knowledge and experience in the fields of chemistry, human factors, process controls, maintenance, procedures, and criticality analysis. The team leader and team members are expected to arrive at the site about noon, Sunday, June 2, 1991.

The licensee has agreed not to resume processing of materials in the solvent extraction system and transfers of solutions to tanks V-103 and V-104 in their interconnecting piping until concurrence is received from the NRC.

The IIT report will constitute the single NRC fact-finding investigation report. It is expected that the team report will be issued within 45 days from now.

Hugh L. Thompson, Jr.

James M. Taylor
Executive Director
for Operations

Enclosure:
As Stated

cc: SECY
OGC
ACRS
GPA
Regional Administrators

ENCLOSURE

# INCIDENT INVESTIGATION TEAM CHARTER

Potential criticality safety problem at General Electric's commercial nuclear fuel manufacturing plant in Wilmington, North Carolina.

The scope of the IIT investigation should include conditions preceding the event; event chronology; systems response, human factors considerations, equipment performance, precursors to the event; emergency response (NRC, licensee, Federal and State agencies); safety significance, radiological considerations; and whether the regulatory process and activities preceding the event contributed to it. Within the framework of this scope, the IIT should specifically:

With respect to conditions preceding the event: Evaluate the activities and plans which established the initial uranium recovery system conditions. Identify the initial uranium recovery system conditions. Identify whether the conditions were prudent and proper. Facts should be obtained regarding the licensee actions associated with providing assurance criticality would not occur. Identify any procedural requirements and/or deficiencies associated with the control of concentration of the waste tank.

With respect to event chronology: Develop and validate a detailed sequence of events associated with and contributing to the inadvertent transfer of approximately 150 kg of uranium to the V-104 waste water treatment tank.

With respect to emergency response: Develop and validate a detailed sequence of events associated with the implementation of the emergency plan implementing procedures including the licensees basis for event classification and the accuracy, timeliness, and effectiveness with which information on the event was reported to the NRC.

With respect to system response: Evaluate the response, reliability, and adequacy of the criticality safety controls of the solvent extraction process, the air spargers, the centrifuge, and the concentration measuring system.

With respect to instrumentation and controls: Evaluate the response and reliability of the digital computer control systems in the uranium recovery system.

With respect to human factors considerations: Evaluate personnel performance, procedures for response to the event, measurement devices and possible errors, and if monitoring to protect workers was adequate.

With respect to the safety significance of the event: Evaluate potential for personnel exposure to radiation or industrial injury, damage to equipment, contamination of equipment, and for material to reach critical mass/configuration.

With respect to the regulatory process and activities preceding the event: Evaluate the adequacy of plant requirements and the adequacy of NRC review of requirements and their implementation to prevent or respond to a potential criticality event.

The scope of the investigation does not include: (1) assessing violations of NRC rules and requirements; (2) reviewing the design or licensing basis for the facility, except as necessary to assess the cause for the event under investigation; and (3) recovery operations.

# APPENDIX B

# INCIDENT SAFETY CONSIDERATIONS

# APPENDIX B

# INCIDENT SAFETY CONSIDERATIONS

## 1 Nuclear Criticality Safety

A nuclear criticality accident involves the uncontrolled fissioning of radioactive material, such as uranium-235. As a result of this fission process, high levels of gamma and neutron radiation are released. The fission process creates radioactive fission products which may escape to the environment, depending on the characteristics of the accident and the degree of containment involved. From past experience, criticality accidents have generally not led to environmental or health and safety consequences outside the facilities in which they occurred. However, levels of radiation in close proximity to the system in which accidental criticalities have occurred have result in fatal radiation doses.

No nuclear criticality occurred as a result of the incident at General Electric (GE) Nuclear Fuel and Components Manufacturing (NFCM). Radiation levels and concentrations of radioactive material in air and effluents remained within the range for normal plant operations. It is beyond the scope of the Incident Investigation Team charter to model the effects of a criticality accident at the licensee's facility. However, the range of radiation levels that might be expected from a nuclear criticality may be computed based on equations in NRC Regulatory Guide 3.34, "Assumption Used for Evaluating Radiological Consequences of Accidental Nuclear Criticality." Hypothetical radiation doses relative to distance for a range of nuclear fissions are shown in Table B.1. Lethal levels of radiation exposure may occur in close proximity (i.e., within 10 feet) for the expected excursion yield of $10^{17}$ fissions, but radiation levels are minimal at distances beyond 1500 feet.

These data indicate that offsite radiological impacts would not be expected from a criticality in the Waste Treatment Facility (WTF) complex for even the highest postulated fission yield of $10^{18}$ fissions. Levels of radiation and radioactive material would be within the limits specified for public protective measures. Consequently, evacuation or protective sheltering of members of the public would not be required. The nearest offsite boundary is located about 1300 feet (approximately 400 meters) from the WTF where the potential criticality incident occurred. The nearest residence is located about 3600 feet (approximately 1100 meters) from the WTF (Figure B.1).

## 2 General Considerations

Since an actual nuclear criticality accident was avoided, the levels of radiation and concentrations of radioactive material in air and effluents were within the range of those measured during routine operations. Thus, there were no offsite consequences in the form of exposures to radiation or radioactive material by members of the public. A review of worker dosimetry, bioassay and whole-body counting results for those individuals involved in the recovery effort showed radiation exposures to be negligible. These exposures are evaluated in Section 3 of this appendix.

During the uranium recovery phase, WTF personnel performed well in a potentially hazardous work environment. Operations, maintenance, health physics and management personnel worked around tankage and systems that contained highly concentrated uranium solutions, acids, and alkaline materials. In addition, WTF personnel and maintenance personnel worked with pressurized air systems (i.e., permanent sparging systems and field-expedient sparging systems), mechanical equipment (pumps and air compressors), and chemical sampling and handling systems. No occupational health and safety accidents or problems occurred as a result of uranium recovery operations with these mechanical and chemical systems.

## 3 External and Internal Radiation Exposure Evaluation

During the recovery phase of the incident, all nonessential personnel working in the proximity of the WTF were evacuated. Access to the WTF was restricted and a check point, approximately one-quarter mile from the WTF, was established by plant security. Only those personnel working on the uranium recovery effort were allowed into the area. These controls were maintained until the quantity of material in the waste tanks was below critical mass limits.

All personnel who worked in the WTF area were provided with self-reading pocket dosimeters (with a range of 0-200 millirem (mr)), in addition to their normal dosimetry. These dosimeters were read and their results recorded at the beginning and end of work activities in the WTF area. A review of the recorded dosimetry results for the period May 30 through June 3, 1991, revealed readings of zero in all but two cases. Both of these were equal to or less than 1 millirem, readings so low that they are difficult to verify.

Both low volume and high volume air samples were taken at the WTF during the incident recovery phase. Low volume air samples (typical flow rates of 120-150 standard cubic feet/hr) were collected, as in normal operations, on each of the eight-hour shifts. A review of the results of sampling data from May 27 through June 2, 1991, for the WTF revealed uranium concentrations in air ranging from $0.3 \times 10^{-11}$ to $2.1 \times 10^{-11}$ $\mu$Ci/cc. High volume air sample results for samples taken on May 30, 1991, showed concentration values ranging from $1.7 \times 10^{-11}$ to $2.7 \times 10^{-11}$ $\mu$Ci/cc. The licensee evaluated "assigned airborne" (AA) concentration exposures for personnel working at the WTF. The AA results for 46 persons involved in recovery activities at the WTF varied from 0 to 4.0 MPC-hours. The regulatory limit for exposure to airborne concentrations is predicated on exposure to 520 MPC-hours.

Bioassays for uranium were also conducted for personnel who worked at the WTF during incident recovery. A review of urinalysis results for the 46 persons who frequented the WTF from May 30 through June 3, 1991, revealed uranium concentration results ranging from 5 (the minimum detectable amount [MDA]) to 12 $\mu$ gram/liter uranium in urine. The licensee's initial action level based on urinalysis results is 15 $\mu$ gram/liter. At this action level, the intake of radioactive material is calculated and if the intake is $\geq$ 9.6 milligram, then work restrictions are imposed. The licensee's urinalysis status report for the period May 30 through June 3, 1991, for the WTF area showed that the uptake of uranium was zero for all personnel sampled. As a further internal exposure control measure, the licensee uses whole-body counting to evaluate exposures to radioactive material via inhalation, ingestion, or absorption pathways. Whole-body counting was conducted for those personnel who worked at the WTF and received an AA value of 1.0 MPC-hour or more. Whole-body count results were generally less than MDA (50 $\mu$ gram U-235). None of the count results exceeded the licensee's initial action point, 150 $\mu$ gram U-235, for increased counting. The licensee's action limit for restricting work in a radioactive materials area is 250 $\mu$ gram of uranium-235.

## 4 Other Radiological Measurements

A review of radiological surveillance data from monitoring programs at the WTF or in close proximity to the WTF revealed no abnormal or unusual levels of radiation during the incident and uranium recovery period (May 28-June 3, 1991). This finding is supported by the data presented below.

Radiation levels in the centrifuge room of the WTF are monitored by use of a thermoluminescent dosimeter (TLD) badge. This badge is used primarily as part of the criticality monitoring program but yields data which can be used to evaluate radiation levels in the WTF area. A review of badge results for the period April 1, 1991, to June 5, 1991, reveals a badge dose of 40 millirem. This exposure level is typical of routine operations.

Nuclear criticality detectors are located at the WTF lagoon system. Detector measurements are transmitted to a console in the Radiation Safety Office (and Emergency Control Center). A review of measurements for the period May 29 through June 1, 1991, showed radiation levels that were consistently well below 0.5 mr/hr.

As part of its environmental radiological surveillance program, the licensee maintains four onsite ambient air sample stations. Data from these air sample stations were reviewed for the period of record, May 23-30, 1991 and May 30-June 6, 1991. During this period, the results ranged from $1.4 \times 10^{-15}$ to $2.9 \times 10^{-15}$ $\mu$Ci/cc. The concentration limits specified in 10 CFR 20, Appendix B, Table II for insoluble Uranium-235 in air is $4 \times 10^{-12}$ $\mu$Ci/cc. Thus, readings during the incident and uranium recovery period were well below regulatory limits.

Table B.1 Hypothetical radiation levels* from an accidental criticality

| Distance (Meters) | Prompt Dose (neutrons + gamma) | | |
|---|---|---|---|
| | $10^{16}$ Fissions | $10^{17}$ Fissions | $10^{18}$ Fissions |
| 10 | 9 rem | 86 rem | 865 rem |
| 100 | .05 rem | 5.7 rem | 6 rem |
| 500 | < 1 mrem | 3.6 mrem | .04 rem |
| 1000 | Background | .1 mrem | 1 mrem |

*Based on information from NRC Regulatory Guide 3.34, "Assumptions Used for Evaluating Radiological Consequences of Accidental Nuclear Criticality."

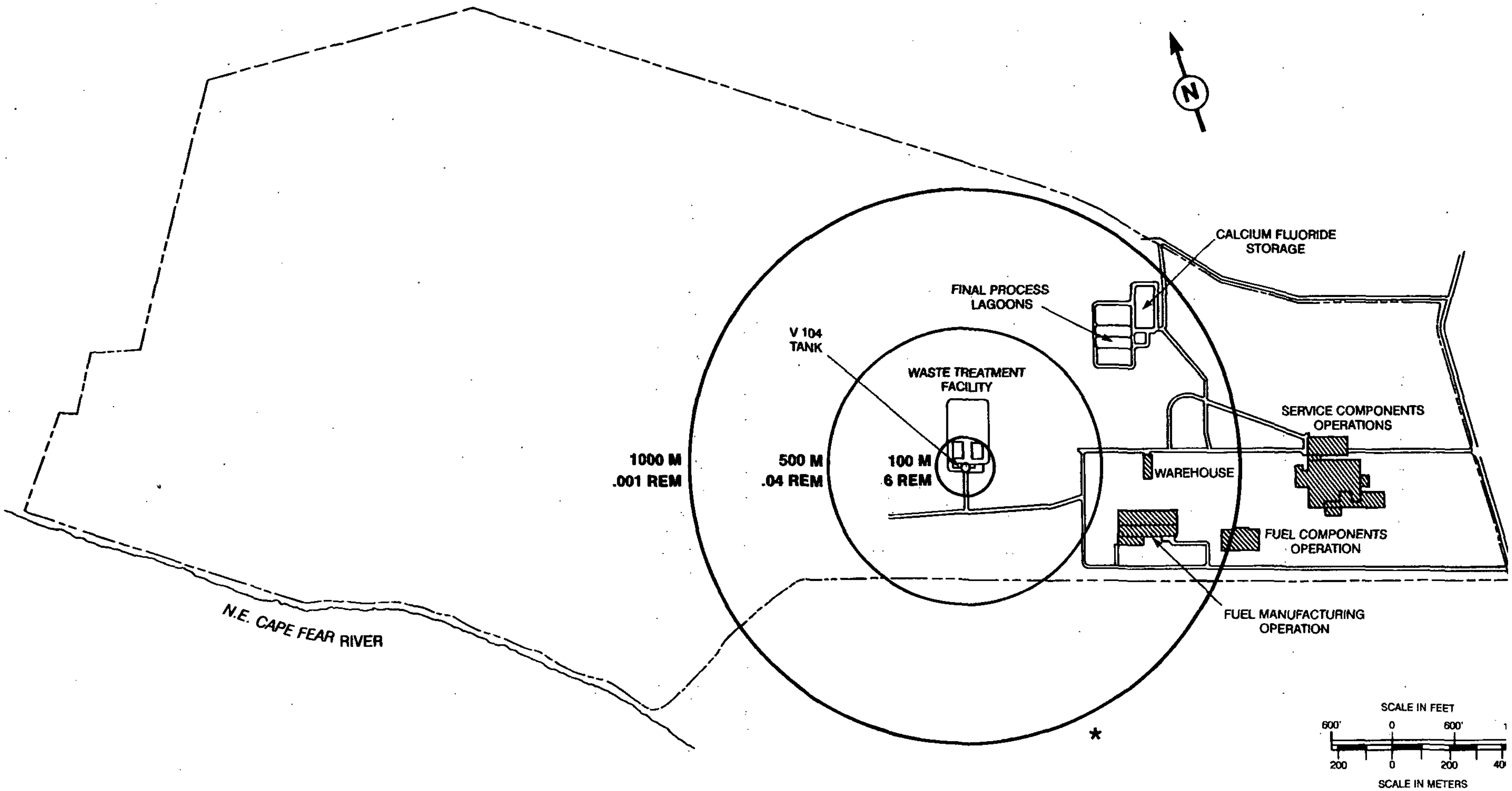



Figure B.1 Radiation doses vs. distances from WTF for hypothetical criticality accident

* Approximate location of nearest residence to site.

# APPENDIX C

# URANIUM HEXAFLUORIDE CONVERSION

# AND

# FUEL FABRICATION PROCESSES

# APPENDIX C

# URANIUM HEXAFLUORIDE CONVERSION

# AND

# FUEL FABRICATION PROCESSES

## 1 Uranium Hexafluoride Conversion

### 1.1 Uranium Hexafluoride to Uranium Dioxide Conversion by the ADU Process

The conversion of uranium hexafluoride ($UF_6$) to uranium dioxide ($UO_2$) powder by water hydrolysis and ammonia precipitation is referred to as the ammonium diuranate (ADU) process. The steps involved in this process are outlined in Figure C.1.

The process starts with the vaporization of $UF_6$. $UF_6$ cylinders are taken from storage by overhead bridge cranes and placed in vaporization chambers. With the cylinder at ambient temperature, the valve cover is removed and flexible tubing (the "pigtail") is connected. The pigtail connection is pressure checked before the cylinder valve is opened and heat is applied. The cylinder is heated within its vaporization chamber by the recirculation of electrically heated air within the closed system. During heating, the cylinder is connected to a pressure sensor which monitors the increase in pressure. Vaporized $UF_6$ flows through the heated piping system to the hydrolysis stage.

In the hydrolysis stage, the vaporized $UF_6$ interacts with water when the vaporized $UF_6$ is introduced beneath the water level in a "safe geometry" hydrolysis receiver tank. A safe geometry hydrolysis scrubber tank, with a recirculating pump and eductor, vents the receiver tank and the hood enclosing the top of the receiver tank. The hydrolysis receiver tank level is an automatic control loop, with a low-level switch interlocked to $UF_6$ flow. The receiver tank also has two high-temperature switches, one in the liquid and one in the hood vent duct, which are interlocked with the $UF_6$ flow. The hydrolysis scrubber tank has a low-level interlock with $UF_6$ flow and a high-conductivity interlock with $UF_6$ flow. The scrubber recirculation pump is also interlocked with $UF_6$ flow. After hydrolysis, the product is sent to a precipitation and dewatering step.

The hydrolysis product is pumped to a precipitation tank where the ADU is precipitated using ammonium hydroxide. The solids are concentrated in a centrifuge, then pumped to the calciner-defluorinator. The liquid may be further treated by a clarifier centrifuge to remove solids, if necessary. The liquid is then pumped to the fluoride waste system. All components of the ADU precipitation and dewatering process are analyzed as geometrically safe.

From the precipitation and dewatering steps, the ADU material is fed as a paste-like material to the defluorinators. The defluorinators are rotating, gas-fired kilns. The ADU paste is continuously fed into the rotating, horizontal kiln chamber where it is dried and reduced to uranium dioxide ($UO_2$). The defluorinators are designed to calcine and transport the $UO_2$ powder within a geometrically safe tube, ultimately discharging the $UO_2$ powder into a geometrically safe discharge hopper and into 5-gallon cans. The offgas from the defluorinator is water scrubbed and discharged into the plant process venting system.

## 1.2 $UF_6$ to $UO_2$ Conversion by the Direct Process

In addition to the ADU process for producing $UO_2$, a direct conversion process may be used. The details of the direct process is considered proprietary by the General Electric Company; however, a general discussion of the process is presented below.

In this process, $UF_6$ cylinders are heated to vaporize the $UF_6$. After $UF_6$ is vaporized, the gas leaves the cylinder through the cylinder valve, passes through a flexible connection (pigtail), and then into the $UF_6$ piping. To prevent solidification of the $UF_6$ gas, all piping and valves are heated. The $UF_6$ pressure from the cylinder is controlled by the heating system. The hot $UF_6$ gas is fed to specially designed chemical reactor vessels. After reaction, the product material is passed through a filter and then to a defluorinator-calciner system where $UO_2$ powder is produced.

# 2 Fuel Fabrication

## 2.1 $UO_2$ Powder Operation

$UO_2$ powder produced in the conversion process undergoes further processing known as pre-treatment and blending.

### 2.1.1 Pre-Treatment

Pre-treatment consists of milling the powder, pressing it into slugs, and then granulating the slugs into powder of a standard particle size. Dry powder in 5-gallon cans is transported to a hammermill. The cans are lifted and inverted to feed the hammermill. This process takes place inside a hood designed to prevent airborne contamination. The hammermill pulverizes the powder utilizing high speed, rotating hammers. The pulverized powder is collected in a hopper. The pulverized powder is moved through a pipe by vibration and a controlled stream of nitrogen to a slab hopper located in the bottom of the baghouse. The nitrogen serves the dual purpose of moving the powder and minimizing oxidation. The powder is then moved into position for pressing into slugs by a mechanical/hydraulic shuttle. The material is then compacted in a die cavity and transferred to a granulator by a slab-type, vibrating conveyor.

The granulator crushes the compacted material into powder of uniform particle size. The powder is then collected in a hopper containing magnets for removal of magnetic impurities.

It then falls by gravity through a hose into a transfer and storage container. A statistical sampling of the container is analyzed for moisture and uranium content. The containers are then transferred to a storage warehouse.

### 2.1.2 $UO_2$ Powder Blending

$UO_2$ powder is blended by use of a slab blender to adjust the uranium enrichment or to assure the homogeneity of physical properties for a batch of $UO_2$ powder.

Powder to be blended is accumulated in 5-gallon cans and transferred to hold areas in the immediate area of the blender. The cans are then transferred, one at a time, to an enclosed and ventilated hood, where the powder is conveyed pneumatically into the slab blender.

After the blender has been charged, it is closed and rotated. Upon completion of the blend, the blended material is discharged pneumatically into 5-gallon cans which are subsequently transferred to a holding area to await enrichment verification.

## 2.2 Pellet Manufacturing

Following the pre-treatment and blending steps, the $UO_2$ powder enters the pellet manufacturing process. This process includes pellet pressing, pellet sintering, and pellet grinding.

The first step in this process is pellet pressing. Containers of $UO_2$ powder are manually positioned, one at a time, in hoods which contain hoppers connected to tubes that feed $UO_2$ powder to the pellet presses on the process floor below. Pellets formed by the pellet presses are approximately 0.5-inch long and 0.5-inch in diameter, and are transferred into permanently numbered, box-shaped containers that are referred to as "furnace boats."

In the sintering process, furnace boats are received in the furnace area and charged into furnaces that are normally maintained at about 1800°C. As the boats pass through the furnaces, the pellets are sintered to about 96 percent of the theoretical $UO_2$ density. Upon discharge from the furnaces, the furnace boats are stored on a slab-roller conveyor system prior to entering the next processing area, which is grinding.

To obtain a uniform pellet diameter, all sintered pellets are processed through mechanical grinders. Pellets to be ground are received in the furnace boats. The pellets are then transferred from the furnace boats to the grinder feeder for grinding. As pellets are ground to the specified diameters, they are inspected for defects and placed in temporary storage trays to await transfer to the rod loading area.

## 2.3 Fuel Rod Loading

Trays of pellets are delivered to the rod-loading station. The fuel pellets are placed in columns within small channels, and weighed and measured to the required specifications. The column of pellets is pushed into an empty Zircaloy tube previously welded at one end with an end plug. The loaded tubes, called fuel rods, are placed in rod trays. Ultimately, individual rods are inserted into a controlled-atmosphere weld box. The inside of the rod is evacuated, then backfilled to positive pressure with inert gas. The end plug is then inserted and welded.

## 2.4 Gadolinia Operations

A separate process is used to produce fuel containing gadolinia, a non-radioactive nuclear "poison" that controls the burn-rate of fuel once it is placed into a reactor. $UO_2$ is received from the powder production area and blended with the required quantity of gadolinia. The process steps are isolated from but follow the same process flow as that described above for $UO_2$ pellets, i.e., pressing, sintering, grinding, and rod loading. Solid waste streams containing gadolinia are processed through solvent extraction and the recovered uranium is returned to the process through the uranyl nitrate conversion process.

## 2.5 Fuel Bundle Assembly

Fuel rods in trays are removed from storage and scanned individually for their U-235 content using an active scanner that employs a neutron source and appropriate gamma radiation detectors. $UO_2$ rods containing gadolinia are scanned for U-235 on a passive scanner.

Based on the rod enrichment requirements of the fuel bundle design, the required rods are removed from the storage trays and transferred to another tray that is used to accumulate all the rods. They are then transferred to an assembly table, cleaned, and visually inspected. These rods are then assembled into a single specific fuel bundle by an automated assembly machine. There are also three additional manual assembly stations. After assembly, the bundle is moved by an overhead crane to the leak test and final inspection station.

## 2.6 Fuel Bundle Storage

Following leak testing, bundles are moved to a bundle storage rack or to a final inspection station where each fuel bundle is wrapped with a plastic dust cover.

The storage area consists of eight rows of racks. Four of these rows are blocked off and are not used in order to allow storage of fuel bundles of higher-than-average enrichment than authorized by the original design. Each rack can hold up to 56 bundles. The racks are rigidly constructed of steel girders on 48-inch centerlines.

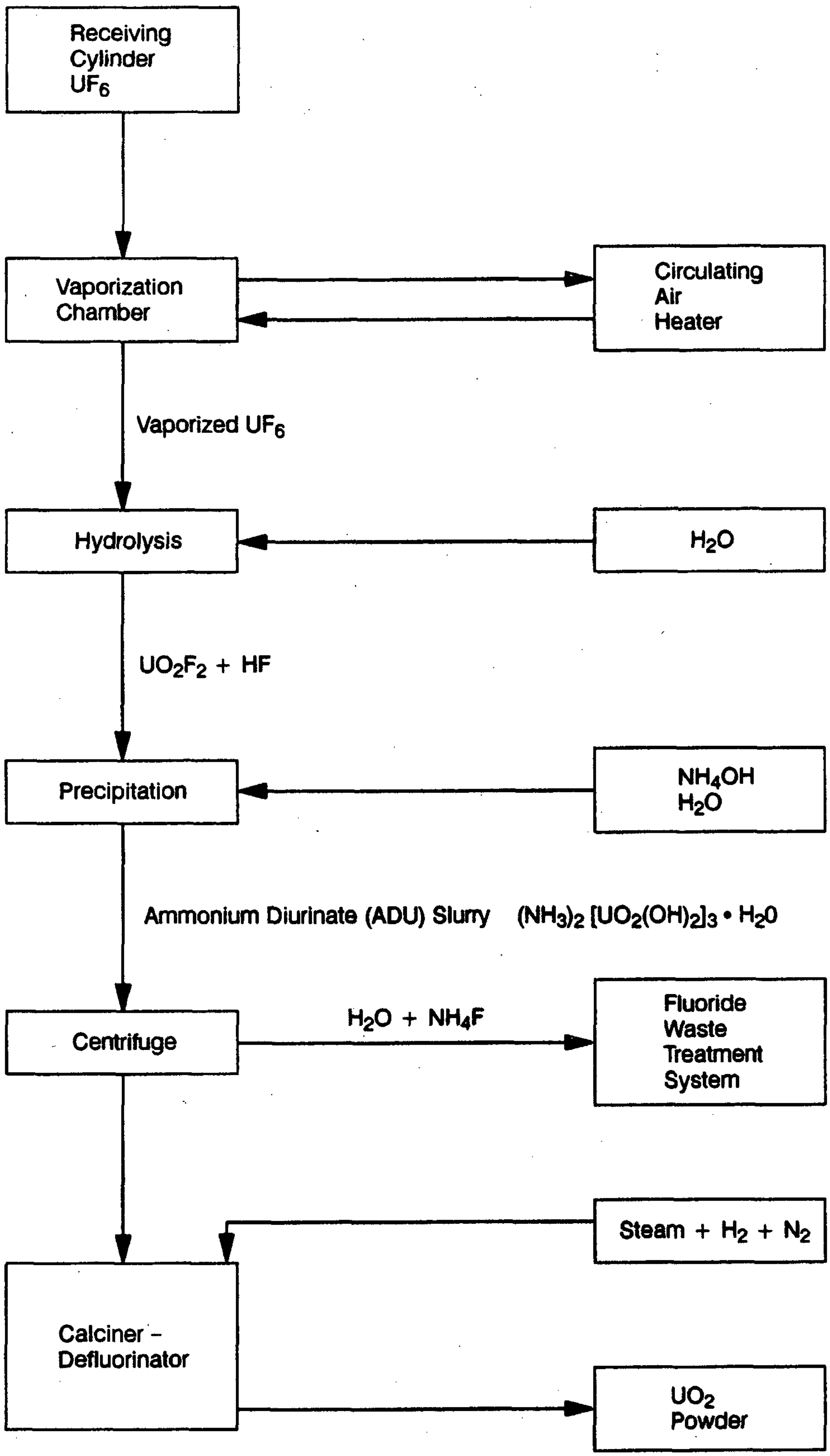



Figure C.1 $UF_6$ to $UO_2$ conversion – ADU process

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At the General Electric Nuclear Fuel and Component Manufacturing facility, located near Wilmington, North Carolina, on May 28 and 29, 1991, approximately 150 kilograms of uranium were inadvertently transferred from safe process tanks to an unsafe tank located at the waste treatment facility, thus creating the potential for a localized criticality safety problem. The excess uranium was ultimately safely recovered when the tank contents were centrifuged to remove the uranium-bearing material. Subsequently, the U.S. Nuclear Regulatory Commission dispatched an Incident Investigation Team to determine what happened, to identify probable causes, and to make appropriate findings and conclusions. This report describes the incident, the methodology used by the team in its investigation, and presents the team's findings and conclusions.

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