NUREG-0090
Vol. 7, No. 4

Report to Congress on Abnormal Occurrences

October - December 1984

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

Available from

Superintendent of Documents
U.S. Government Printing Office
Post Office Box 37082
Washington, D.C. 20013-7082

A year's subscription consists of 4 issues for this publication.

Single copies of this publication are available from National Technical Information Service, Springfield, VA 22161

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Washington, D.C. 20555

Previous Reports in Series

NUREG 75/090, January-June 1975,
published October 1975

NUREG-0090-1, July-September 1975,
published March 1976

NUREG-0090-2, October-December 1975,
published March 1976

NUREG-0090-3, January-March 1976,
published July 1976

NUREG-0090-4, April-June 1976,
published March 1977

NUREG-0090-5, July-September 1976,
published March 1977

NUREG-0090-6, October-December 1976,
published June 1977

NUREG-0090-7, January-March 1977,
published June 1977

NUREG-0090-8, April-June 1977,
published September 1977

NUREG-0090-9, July-September 1977,
published November 1977

NUREG-0090-10, October-December 1977,
published March 1978

NUREG-0090, Vol.1, No.1, January-March 1978,
published June 1978

NUREG-0090, Vol.1, No.2, April-June 1978,
published September 1978

NUREG-0090, Vol.1, No.3, July-September 1978,
published December 1978

NUREG-0090, Vol.1, No.4, October-December 1978,
published March 1979

NUREG-0090, Vol.2, No.1, January-March 1979,
published July 1979

NUREG-0090, Vol.2, No.2, April-June 1979,
published November 1979

NUREG-0090, Vol.2, No.3, July-September 1979,
published February 1980

NUREG-0090, Vol.2, No.4, October-December 1979,
published April 1980

NUREG-0090, Vol.3, No.1, January-March 1980,
published September 1980

NUREG-0090, Vol.3, No.2, April-June 1980,
published November 1980

NUREG-0090, Vol.3, No.3, July-September 1980,
published February 1981

NUREG-0090, Vol.3, No.4, October-December 1980,
published May 1981

NUREG-0090, Vol.4, No.1, January-March 1981,
published July 1981

NUREG-0090, Vol.4, No.2, April-June 1981,
published October 1981

NUREG-0090, Vol.4, No.3, July-September 1981,
published January 1982

NUREG-0090, Vol.4, No.4, October-December 1981,
published May 1982

NUREG-0090, Vol.5, No.1, January-March 1982
published August 1982

NUREG-0090, Vol.5, No.2, April-June 1982,
published December 1982

NUREG-0090, Vol.5, No.3, July-September 1982,
published January 1983

NUREG-0090, Vol.5, No.4, October-December 1982,
published May 1983

NUREG-0090, Vol.6, No.1, January-March 1983,
published September 1983

NUREG-0090, Vol.6, No.2, April-June 1983,
published November 1983

NUREG-0090, Vol.6, No.3, July-September 1983,
published April 1984

NUREG-0090, Vol.6, No.4, October-December 1983,
published May 1984

NUREG-0090, Vol.7, No.1, January-March 1984,
published July 1984

NUREG-0090, Vol.7, No.2, April-June 1984,
published October 1984

NUREG-0090, Vol.7, No.3, July-September 1984,
published April 1985

ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from October 1 to December 31, 1984.

The report states that for this report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved four control rods failing to insert during testing and the other involved degraded upper head injection system accumulator isolation valves. There was one abnormal occurrence at a fuel cycle facility; the event involved buildup of uranium in a ventilation system. There was one abnormal occurrence reported by an Agreement State; the event involved an overexposure of a radiographer trainee.

The report also contains information updating some previously reported abnormal occurrences.

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PREFACE

INTRODUCTION

The Nuclear Regulatory Commission reports to the Congress each quarter under provisions of Section 208 of the Energy Reorganization Act of 1974 on any abnormal occurrences involving facilities and activities regulated by the NRC. An abnormal occurrence is defined in Section 208 as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety.

Events are currently identified as abnormal occurrences for this report by the NRC using the criteria delineated in Appendix A. These criteria were promulgated in an NRC policy statement which was published in the Federal Register on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952). In order to provide wide dissemination of information to the public, a Federal Register notice is issued on each abnormal occurrence with copies distributed to the NRC Public Document Room and all local public document rooms. At a minimum, each such notice contains the date and place of the occurrence and describes its nature and probable consequences.

The NRC has reviewed Licensee Event Reports, licensing and enforcement actions (e.g., notices of violations, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report covers the period from October 1 to December 31, 1984.

Information reported on each event includes: date and place; nature and probable consequences; cause or causes; and actions taken to prevent recurrence.

THE REGULATORY SYSTEM

The system of licensing and regulation by which NRC carries out its responsibilities is implemented through rules and regulations in Title 10 of the Code of Federal Regulations. To accomplish its objectives, NRC regularly conducts licensing proceedings, inspection and enforcement activities, evaluation of operating experience and confirmatory research, while maintaining programs for establishing standards and issuing technical reviews and studies. The NRC's role in regulating represents a complete cycle, with the NRC establishing standards and rules; issuing licenses and permits; inspecting for compliance; enforcing license requirements; and carrying on continuing evaluations, studies and research projects to improve both the regulatory process and the protection of the public health and safety. Public participation is an element of the regulatory process.

In the licensing and regulation of nuclear power plants, the NRC follows the philosophy that the health and safety of the public are best assured through the establishment of multiple levels of protection. These multiple levels can be achieved and maintained through regulations which specify requirements which will assure the safe use of nuclear materials. The regulations include design and quality assurance criteria appropriate for the various activities licensed by NRC. An inspection and enforcement program helps assure compliance with the regulations.

Most NRC licensee employees who work with or in the vicinity of radioactive materials are required to utilize personnel monitoring devices such as film badges or TLD (thermoluminescent dosimeter) badges. These badges are processed periodically and the exposure results normally serve as the official and legal record of the extent of personnel exposure to radiation during the period the badge was worn. If an individual's past exposure history is known and has been sufficiently low, NRC regulations permit an individual in a restricted area to receive up to three rems of whole body exposure in a calendar quarter. Higher values are permitted to the extremities or skin of the whole body. For unrestricted areas, permissible levels of radiation are considerably smaller. Permissible doses for restricted areas and unrestricted areas are stated in 10 CFR Part 20. In any case, the NRC's policy is to maintain radiation exposures to levels as low as reasonably achievable.

REPORTABLE OCCURRENCES

Actual operating experience is an essential input to the regulatory process for assuring that licensed activities are conducted safely. Reporting requirements exist which require that licensees report certain incidents or events to the NRC. This reporting helps to identify deficiencies early and to assure that corrective actions are taken to prevent recurrence.

For nuclear power plants, dedicated groups have been formed both by the NRC and by the nuclear power industry for the detailed review of operating experience to help identify safety concerns early, to improve dissemination of such information, and to feed back the experience into licensing, regulations, and operations.

In addition, the NRC and the nuclear power industry have ongoing efforts to improve the operational data system which include not only the type, and quality, of reports required to be submitted, but also the methods used to analyze the data. Two primary sources of operational data are reports submitted by the licensees under the Licensee Event Report (LER) system, and under the Nuclear Plant Reliability Data (NPRD) system. The former system is under the control of the NRC while the latter system is a voluntary, industry-supported system operated by the Institute of Nuclear Power Operations (INPO), a nuclear utility organization.

Some form of LER reporting system has been in existence since the first nuclear power plant was licensed. Reporting requirements were delineated in the Code of Federal Regulations (10 CFR), in the licensees' technical specifications, and/or in license provisions. In order to more effectively collect, collate, store, retrieve, and evaluate the information concerning reportable events, the Atomic Energy Commission (the predecessor of the NRC) established in 1973 a

computer-based data file, with data extracted from licensee reports dating from 1969. Periodically, changes were made to improve both the effectiveness of data processing and the quality of reports required to be submitted by the licensees.

Effective January 1, 1984, major changes were made to the requirements to report to the NRC. A revised Licensee Event Report System (10 CFR § 50.73) was established by Commission rulemaking which modified and codified the former LER system. The purpose was to standardize the reporting requirements for all nuclear power plant licensees and eliminate reporting of events which were of low individual significance, while requiring more thorough documentation and analyses by the licensees of any events required to be reported. All such reports are to be submitted within 30 days of discovery. The revised system also permits licensees to use the LER procedures for various other reports required under specific sections of 10 CFR Part 20 and Part 50. The amendment to the Commission's regulations was published in the Federal Register (48 FR 33850) on July 26, 1983, and is described in NUREG-1022, "Licensee Event Report System," and Supplement 1 to NUREG-1022.

Also effective January 1, 1984, the NRC amended its immediate notification requirements of significant events at operating nuclear power reactors (10 CFR § 50.72). This was published in the Federal Register (48 FR 39039) on August 29, 1983, with corrections (48 FR 40882) published on September 12, 1983. Among the changes made were the use of terminology, phrasing, and reporting thresholds that are similar to those of 10 CFR § 50.73. Therefore, most events reported under 10 CFR § 50.72 will also require an in-depth follow-up report under 10 CFR § 50.73.

The NPRD system is a voluntary program for the reporting of reliability data by nuclear power plant licensees. Both engineering and failure data are to be submitted by licensees for specified plant components and systems. In the past, industry participation in the NPRD system was limited and, as a result, the Commission considered it may be necessary to make participation mandatory in order to make the system a viable tool in analyzing operating experience. However, on June 8, 1981, INPO announced that because of its role as an active user of NPRD system data, it would assume responsibility for management and funding of the NPRD system. INPO reports that significant improvements in licensee participation are being made. The Commission considers the NPRD system to be a vital adjunct to the LER system for the collection, review, and feedback of operational experience; therefore, the Commission periodically monitors the progress made on improving the NPRD system.

Information concerning reportable occurrences at facilities licensed or otherwise regulated by the NRC is routinely disseminated by the NRC to the nuclear industry, the public, and other interested groups as these events occur.

Dissemination includes special notifications to licensees and other affected or interested groups, and public announcements. In addition, information on reportable events is routinely sent to the NRC's more than 100 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

The Congress is routinely kept informed of reportable events occurring at licensed facilities.

AGREEMENT STATES

Section 274 of the Atomic Energy Act, as amended, authorizes the Commission to enter into agreements with States whereby the Commission relinquishes and the States assume regulatory authority over byproduct, source and special nuclear materials (in quantities not capable of sustaining a chain reaction). Comparable and compatible programs are the basis for agreements.

Presently, information on reportable occurrences in Agreement State licensed activities is publicly available at the State level. Certain information is also provided to the NRC under exchange of information provisions in the agreements. NRC prepares a semiannual summary of this and other information in a document entitled, "Licensing Statistics and Other Data," which is publicly available.

In early 1977, the Commission determined that abnormal occurrences happening at facilities of Agreement State licensees should be included in the quarterly report to Congress. The abnormal occurrence criteria included in Appendix A are applied uniformly to events at NRC and Agreement State licensee facilities. Procedures have been developed and implemented and abnormal occurrences reported by the Agreement States to the NRC are included in these quarterly reports to Congress.

FOREIGN INFORMATION

The NRC participates in an exchange of information with various foreign governments which have nuclear facilities. This foreign information is reviewed and considered in the NRC's assessment of operating experience and in its research and regulatory activities. Reference to foreign information may occasionally be made in these quarterly abnormal occurrence reports to Congress; however, only domestic abnormal occurrences are reported.

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

OCTOBFR - DECEMBER 1984

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the fourth calendar quarter of 1984. As of the date of this report, the NRC had determined that the following were abnormal occurrences.

84-17 Four Control Rods Fail to Insert During Testing

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see general criterion No. 2) of this report notes that a major degradation of essential safety-related equipment can be considered an abnormal occurrence.

Date and Place - On October 6, 1984, during quarterly individual control rod scram testing for Susquehanna Unit 1, four rods failed to insert. Nine other rods hesitated before scramming; however, they did fully scram within the time allowed by the plant's technical specifications. Susquehanna Unit 1, which utilizes a General Electric (GE)-designed boiling water reactor (BWR), is operated by the Pennsylvania Power and Light Company (PP&L). The plant is located in Luzerne County, Pennsylvania.

Nature and Probable Consequences - GE-designed BWRs utilize control rods driven in from the bottom of the core. During a reactor scram, the rods must be rapidly inserted into the core against the force of gravity. To accomplish this, as well as normal withdrawals and insertions of control rods, each control rod drive (each control rod has its own drive) is operated by a double acting piston which moves the control rod in and out of the core by a hydraulic system which provides water under pressure to operate the piston. To withdraw a rod, water under pressure is admitted to the area above the piston (to provide the motive force), and water under the piston exits to an exhaust header. The opposite takes place to insert a rod. During normal rod insertion or withdrawal, the system is designed to move the rod relatively slowly and movement is limited to a short distance of travel.

During a scram, a separate set of valves functions to effect rod movement. Each control rod drive has a scram inlet valve and a scram outlet valve. At Susquehanna, these valves are normally held closed during reactor operation by instrument air pressure supplied by their T-ASCO (Automatic Switch Company) scram pilot solenoid valve (SPSV). There are 185 SPSVs installed (one per control rod drive). The SPSVs are energized by the reactor protection system (RPS). Upon receipt of a scram signal from the RPS, the SPSVs deenergize which rapidly vents the air pressure from the scram inlet and outlet valves, allowing them to open and the rod to scram.

Opening the scram inlet valve permits high pressure water to the area below the drive piston. Opening the scram outlet valve vents the area over the drive piston to the scram discharge volume. The large differential pressure across the piston produces a large upper force on the control rod, giving it a high acceleration and providing a high margin of force to overcome possible friction within the control rod drive.

For surveillance testing purposes, test switches are provided which permit each control rod to be scrammed individually, rather than all rods scramming upon receipt of a scram signal from the RPS. The test switches permit control rod testing, as required periodically by the technical specifications, without shutting the plant down.

On October 6, 1984, with Unit 1 at 60% power, quarterly individual rod scram testing began on ten percent of the rods (19) as required by technical specifications. Control rod 42-23 failed to insert; the test was repeated three times and each time the rod failed to scram. Instrument and Control technicians investigated the problem and found that when the rod's SPSV was physically struck, the rod scrammed. The rod was then fully withdrawn and retested; this time the rod scrammed satisfactorily. When rod testing continued, rod 42-39 also failed to scram. Similar to rod 42-23, when the SPSV for rod 42-39 was struck, the rod scrammed. The licensee then decided to individually scram the rest of the 185 rods. Two additional rods (i.e., 58-31 and 38-39) failed to scram; again, they did scram when their SPSVs were struck. Both were individually fully withdrawn, and on retest, scrammed satisfactorily. Nine other rods hesitated initially when tested, but did meet the required maximum insertion time of seven seconds.

On October 7, the SPSVs were replaced on the four rods which failed to scram. The licensee set up a task force to investigate the SPSV failures. One of the failed SPSVs was sent to GE, San Jose, California, and another was sent to Franklin Research Center for analysis of the failure mechanism.

On October 12, 1984, GE informed PP&L that the SPSV failed due to the disc holder subassembly disc sticking to the seat on the valve body. The sticking was due to a degradation of the polyurethane disc material. GE, as a product upgrade, had changed the disc material in new SPSVs to Viton-A a few years ago. The Viton-A material was subsequently incorporated into ASCO spare part kits for these valves, beginning in 1982.

Based on this information and the determination that most if not all of the SPSVs on both units were affected, the licensee decided to shut down both Units 1 and 2. Shutdown began on October 12; both units were in hot shutdown on October 13. Unit 2 subsequently went to cold shutdown the same day to conduct unrelated maintenance. The NRC headquarters operations center was notified of these shutdowns by the Emergency Notification System (ENS).

ASCO spare part kits containing the Viton-A disc were obtained by the licensee and the disc holder subassemblies were extracted from the spare part kits and installed in all 185 SPSVs on Unit 1 during October 13 through October 15, 1984. These valves were functionally tested by observing the scram inlet and outlet valves stroking when the individual rod test switches were actuated. Additionally, individual rod testing was performed while shut down. On Unit 2, the licensee determined that 93 of the SPSVs had been previously rebuilt in April

1983 using ASCO spare part kits containing the Viton-A disc. The licensee installed the new disc subassembly in the remaining 92 SPSVs and also inspected the ones that were previously rebuilt to ensure that they contained the Viton-A disc.

For added reliability to the reactor scram process, the CRD instrument air system has two DC solenoid operated, three-way air valves (called backup scram valves) installed on the supply header. These valves operate, similar to the SPSVs, upon receipt of a signal from the RPS. Upon energization, either of the two backup scram valves can vent the entire CRD instrument air system. The air supplied to the hydraulic control units of the CRDs, and the scram discharge volume (SDV) vent and drain pilot valves, passes through these two backup scram valves. Therefore, these two valves provide backup scram capability to the individual SPSVs and the SDV vent and drain valves.

The backup scram valves and the SDV vent and drain T-ASCO pilot valves on Unit 1 were also rebuilt with new Viton-A discs. Subsequently, on October 17, Unit 1 returned to power and individual rod testing on all rods was conducted at approximately 50 percent power.

On Unit 2, new disc holder subassemblies were also installed in the backup scram valves. The Unit 2 SDV vent and drain pilot valves, manufactured by Valcor, were not affected.

On October 18, 1984, while continuing individual rod scram testing on Unit 1 at about 55% power, the licensee discovered that due to an administrative error, operability testing of the SDV vent and drain valves, required each 18 months by technical specifications, was overdue by about 15 months. Per technical specifications, these valves must close within 30 seconds after receipt of a signal to scram from the RPS. The licensee immediately declared the SDV system inoperable and notified the NRC Operations Center by the ENS. When the licensee could find no documentation from previous reactor scrams that the vent and drain valves had operated properly, the plant was manually scrammed on October 18, 1984. During the scram, the SDV vent valve closed in 32.4 seconds and the drain valve in 26.9 seconds. Since the vent valve did not meet the acceptance criteria of 30 seconds, the SDV remained inoperable while the licensee investigated the cause. To correct the problem, the licensee replaced the apparently undersized T-ASCO pilot valve with the larger Valcor valve, similar to that used on Unit 2. This larger valve vents the air header significantly faster than the smaller T-ASCO valve. Shutdown testing of the Valcor valve following its installation indicated a vent valve closure time of about six seconds; this is similar to the time obtained on Unit 2 during the unit's preoperational testing. Subsequent testing on October 21, 1984, by manually scramming Unit 1 from 7% power, showed a vent valve closure time of 5.2 seconds.

As discussed later, NRC Resident Inspectors performed a special inspection from October 13-22, 1984. One finding noted that six control rods in Unit 1 had experienced hesitation at the initiation of a scram one or more times during full core scrams as far back as March 22, 1983. One of the rods (58-31), which failed to scram during its individual scram test on October 6, 1984, had hesitated on full core scrams on March 22, 1983, June 13, 1984, and July 3, 1984. Control rod 54-47, which hesitated (but did scram) on October 6, 1984, had also hesitated on June 13, 1984, July 3, 1984, and July 15, 1984. The other three rods (i.e., 42-23, 42-39, and 38-39) which failed to scram on October 6, 1984, had also hesitated during the full core scram on June 13, 1984.

A more significant finding of the NRC inspection was that during the June 13, 1984, full core scram, the four rod array containing control rods 38-39, 38-43, 42-39, and 42-43, exceeded the technical specification allowable average scram insertion time from the fully withdrawn position (notch 48) to notch 45. The two slowest rods (i.e., 38-39 and 42-39) of the four rod array were two of the four which failed to insert on October 6, 1984; this is a precursor to the October 6, 1984, event. Even though the computer printouts of the June 13, 1984 scram data had specifically indicated that this rod array exceeded technical specification average scram insertion time, the licensee failed to note this when the printouts were reviewed during control rod surveillance scram testing on June 25, 1984. This discrepancy was missed both by the individual performing the surveillance and by the supervisor who reviewed the completed surveillance.

The safety significance of the October 6, 1984 event was the reduction in the required "extremely high probability" of shutting down the reactor in the event of an anticipated operational occurrence. This is evidenced by the following:

1. During single rod scram testing, four control rods failed to insert, and nine others hesitated before scramming, due to a common mode failure of the SPSVs. There was a potential that the common mode failure aspect could have caused a significant number of control rods to be inoperable. The mechanism that could have possibly identified the problem earlier, the surveillance procedure, was not properly reviewed, and therefore the precursor event on June 13 was not recognized and investigated.

2. Even though the plant has backup scram valves, at the time of the event the condition of the valves was not known since they had not been tested since before the plant originally started up. During the preoperational testing of Unit 1, the time to depressurize the air header for each backup scram valve was 43.3 seconds and 28.21 seconds, respectively. The backup scram valves are not included in technical specification required surveillance testing. In response to an unrelated issue, however, the licensee intends to test these valves on a refueling interval basis although they have not yet been retested since the preoperational test program.

<u>Cause or Causes</u> - The SPSVs failed due to the disc holder subassembly disc sticking to the seat on the T-ASCO valve bodies. The cause of the failure was initially determined to be due to contamination of the polyurethane seat material by oil and/or water which had been introduced into the CRD instrument air system. PP&L is continuing its investigation to determine the exact nature and source/origin of the contaminants found in the instrument air system. The Viton-A replacement material is resistant to all oils which could be introduced into the instrument air system as well as to water and other chemical contaminants.

A contributing cause was the licensee's inadequate review of the data associated with the June 13, 1984, full core scram during the surveillance conducted on June 25, 1984. The data provided information by which the deficiency may have been identified, before some rods actually failed to insert.

<u>Actions Taken to Prevent Recurrence</u>

<u>Licensee</u> - The licensee conformed to the actions contained in the NRC Region 1 Confirmatory Action Letter, dated October 17, 1984 (Ref. 1), discussed below.

The licensee is continuing its investigation to determine the exact nature and source/origin of the contaminants found in the CRD instrument air system. Surveillances and shutdowns, conducted since the various plant modifications were made, have shown the reactor scram systems have performed satisfactorily on both Units 1 and 2. The licensee's responses to the NRC Region I Confirmatory Action Letter are contained in letters dated November 19, 1984 (Ref. 2) and January 9, 1985 (Ref. 3).

After the licensee discovered the administrative error which resulted in the SDV vent and drain valve 18-month operability test being overdue by about 15 months, the licensee conducted a 100 percent documentation review to ensure that no other similar administrative errors were present in their surveillance tracking system. No other deficiencies were found. The licensee also modified the surveillance documentation forms to emphasize the date on which the surveillance was performed rather than the date of the form and created a full time surveillance documentation auditor position. These actions are intended to reduce the potential for incorrect data entries in the surveillance tracking computer system.

NRC - As previously mentioned, NRC headquarters was notified by the licensee, via an ENS call on October 12, 1984, of the defective SPSVs and the licensee's decision to shut down Units 1 and 2. On October 15, 1984, at the request of the NRC Region I, the licensee committed to remain below 5% power pending the results of a meeting in Bethesda, Maryland on the following day to discuss the SPSV problem. At the meeting, the licensee committed to the following actions:

a. Scram-time test all 185 rods, on each unit, when a 50 - 60% power level is reached,

b. Develop a surveillance procedure to unambiguously assess scram pilot valve operability, to be submitted to and approved by NRC prior to implementation, and performed every four to six weeks,

c. Trend and report immediately to NRC, via the ENS network, any failures or anomalies found during scram solenoid valve operability tests, or individual control rod scram time testing (normally performed for a 10% rod sample every four months), and

d. Provide the failure analysis results from Franklin Research Center and General Electric testing on the original valves which failed.

On October 17, 1984, NRC Region I issued a Confirmatory Action Letter confirming the above commitments (Ref. 1).

From October 13 to 22, 1984, a special safety inspection was performed by the NRC Resident Inspectors of the circumstances involved with the failure of the four SPSVs during individual rod scram testing on Unit 1 on October 6, 1984. The inspection consisted of a review and evaluation of: SPSV function, licensee actions following identification of the SPSV failures, scram time surveillance testing, SPSV maintenance history and Unit 1 SDV vent and drain pilot valve inoperability. Some of the findings have been discussed above under "Nature and Probable Consequences." The inspection results were forwarded to the licensee in a letter dated November 15, 1984 (Ref. 4).

An enforcement conference was held at NRC Region I on November 30, 1984, between NRC and licensee personnel to discuss the results of the NRC special safety inspection, and the status of the licensee's actions associated with the NRC Region I Confirmatory Action Letter. Conference details were forwarded to the licensee in an NRC Region I letter dated January 10, 1985 (Ref. 5). NRC Region I forwarded a Notice of Violation to the licensee on January 4, 1985 (Ref. 6). The violation involved the failure of the licensee to recognize, during a control rod scramming surveillance test on June 25, 1984, that technical specification requirements were violated for average scram insertion time to notch position 45 by one, four rod 2 x 2 array; since the licensee failed to discover the violation the reactor was allowed to operate without either repairing these rods or declaring them inoperable and performing required analyses.

The NRC will continue to follow the licensee's actions, as necessary, to assure that they are satisfactory.

Editor's Note

On April 3, 1985, the NRC forwarded Inspection and Enforcement Information Notice No. 85-27 to all nuclear power reactor facilities holding an operating license or a construction permit (Ref. 7). The notice was issued to clarify the requirements for licensees to report to the NRC, by the Emergency Notification System (ENS) and by a Licensee Event Report (LER), an event or condition that results in or could result in multiple failures in safety systems. This clarification was considered necessary since PP&L did not believe it necessary to report the October 6, 1984, failures at Susquehanna Unit 1 to the NRC either by the ENS or by an LER.

Unless new significant information becomes available, this incident is closed for purposes of this report.

* * * * * * * *

84-18 Degraded Upper Head Injection System Accumulator Isolation Valves

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 4 of "For Commercial Nuclear Power Plants") of this report notes that discovery of a major condition not specifically considered in the safety analysis report (SAR) or technical specifications that requires immediate remedial action can be considered an abnormal occurrence.

Date and Place - On November 1, 1984, the upper head injection (UHI) system accumulator isolation valves were discovered to have been incapable of required automatic closure for Duke Power Company's McGuire Nuclear Station, Unit 1, a pressurized water reactor plant, located in Mecklenburg County, North Carolina.

Nature and Probable Consequences - At McGuire Unit 1, the UHI system is an engineered safety feature, designed to provide cooling (borated water) of the core during the blowdown portion of the postulated loss of coolant accident (LOCA) transient for a large rupture in the cold leg of the reactor coolant system (RCS). The system (see Figure 1) consists primarily of two pressure

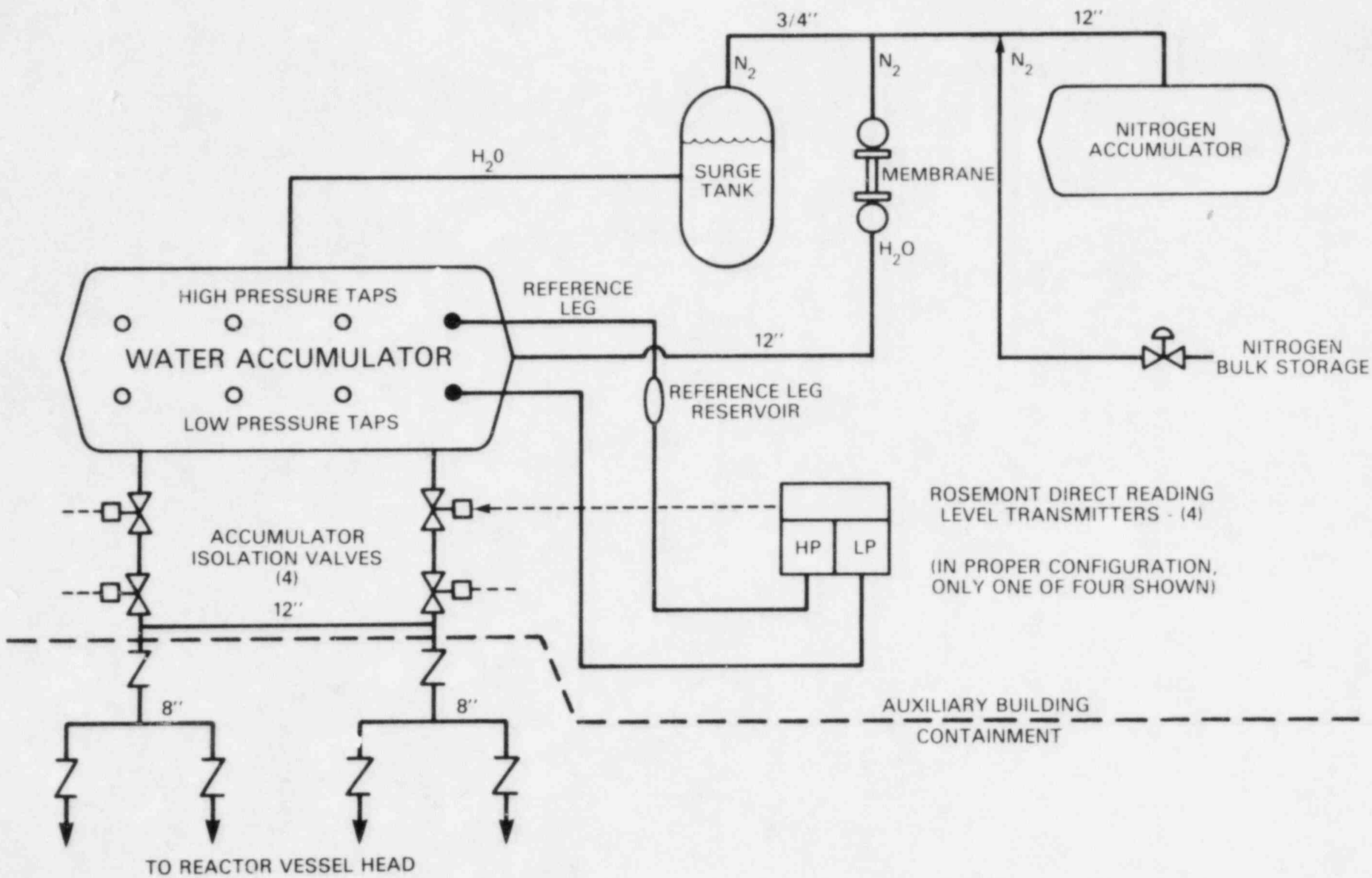

Figure 1 Simplified Piping and Instrumentation Diagram (McGuire Upper Head Injection System)

vessels (accumulators), one filled with borated water and the other with pressurized nitrogen gas. Pressure is maintained to the borated water in the first accumulator by the nitrogen gas in the other accumulator. During normal operations, the contents of the two accumulators are separated by a membrane in the 12" diameter line connecting the accumulators, and pressure is maintained at equilibrium through a surge tank.

Two separate water lines are connected near the bottom of the water accumulator. In each line, there are two accumulator isolation valves and a swing disc check valve in series. The accumulator isolation valves are motor operated and are normally open. Downstream of the check valve, each water line feeds two injection lines. Each injection line contains a swing disc check valve and is connected to the upper head of the reactor vessel. During normal operation, the check valves isolate the UHI system from the RCS.

In the unlikely event of a LOCA large enough to depressurize the RCS below about 1250 psig, the RCS pressure falls below the water accumulator pressure and the borated water is forced through the check valves into the reactor vessel head by the nitrogen gas. When the water level falls to a predetermined level in the accumulator, differential pressure transmitters (which sense accumulator water level) provide an initiating signal for the four isolation valves to close; this is to prevent injection of the nitrogen gas into the RCS.

On October 31, 1984, while Unit 1 was operating at 100% power, the licensee found that nitrogen in excess of technical specifications was entrained in the water accumulator. A plant shutdown was initiated. On November 1, 1984, while the licensee was draining the tank, it was discovered that the four isolation valves failed to close on accumulator water low level. Investigation showed that the valves had been incapable of required automatic closure since April 25, 1984. From this date until the condition was discovered on November 1, 1984, the plant had been operated for about five months. During this period, had a large LOCA occurred, a considerable amount of nitrogen could have been injected into the reactor vessel upper head.

Although the effects of injecting the non-condensable gas has not been analyzed in detail, it could interfere with cooling the reactor core during such an accident. This condition is beyond the design bases for the plant and is not specifically analyzed in the safety analysis report.

Cause or Causes - Investigations revealed that the water accumulator differential pressure transmitters, which sense accumulator water level and provide the initiating signal for isolation valve closure, had been improperly installed on Unit 1. The impulse lines were not connected to the appropriate transmitter ports. This resulted in a loss of function of the transmitters and, consequently, in the inability for automatic closure of the accumulator isolation valves. Further investigation revealed that the transmitters had been incorrectly installed during a plant modification in April 1984, which replaced the Barton differential pressure instruments by Rosemont instruments. The cause of this incorrect installation is attributed to inadequate instructions which did not provide sufficient direction for proper connection of the transmitters. The installation errors were similar to those previously addressed in NRC Inspection and Enforcement Information Notice No. 84-45, which was issued on June 11, 1984 (Ref. 8).

In addition, the functional testing of the system following completion of this modification was limited to a dry calibration of the differential pressure transmitters. This dry calibration was not an adequate method of functional testing because it was unable to detect improper installation of the differential pressure transmitters and did not demonstrate that the transmitters would function properly with respect to water level in the accumulator.

Actions Taken to Prevent Recurrence

Licensee - A new installation procedure has been issued which now requires verification of proper tubing connections for differential pressure transmitters, and the licensee has committed to strengthening the post modification testing program. Additionally, the licensee reviewed other safety-related differential pressure applications for similar problems. No other problems were identified. Prior to startup of Unit 1, the UHI differential pressure transmitters were properly connected and the system functionally tested using an adequate method. Unit 2 was inspected and found to have the UHI differential pressure transmitters properly connected.

NRC - An inspector from NRC Region II was sent to the site on November 2, 1984, to participate in the investigation of the event. All plants which have UHI systems were determined to be located within NRC Region II, and were notified of this problem. Each licensee reported that the configuration of UHI differential pressure transmitters had been inspected and confirmed to be correct. Inspection and Enforcement Information Notice No. 85-02 (Ref. 9), was sent on January 11, 1985, to all reactor facilities with operating licenses or construction permits to alert them of possible problems associated with improper installation of differential pressure transmitters and inadequate post-modification functional testing.

As a result of the NRC Region II inspection on November 2 - 3, 1984, failures to comply with NRC regulatory requirements were identified. An enforcement conference to discuss these matters was held with the licensee at the NRC Region II Office on November 14, 1984. On February 20, 1985, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $50,000 (Ref. 10). The forwarding letter also included NRC Inspection Report Nos. 50-369/84-34 and 50-370/84-31.

The NRC Office of Nuclear Reactor Regulation (NRR) is currently attempting to assess the effects of the accumulator isolation valves failing to close during a large LOCA in a plant with a UHI system. NRR is also considering initiation of additional studies regarding the net safety benefit of the UHI system and changes in the technical specification requirements. It is also noted that the licensee is investigating the efficacy of removal of the UHI system.

Editor's Note

The licensee has experienced other problems with the UHI system. For example, during corrective actions associated with the degraded accumulator isolation valves, the licensee discovered that the Unit 1 accumulator differential pressure instrument trip set points had been set incorrectly since March 1983. This condition could have resulted in charging over 3000 gallons less than the prescribed quantity of borated water for UHI injection under accident conditions. This deficiency would not have had an effect on UHI actuation following the

In addition, the functional testing of the system following completion of this modification was limited to a dry calibration of the differential pressure transmitters. This dry calibration was not an adequate method of functional testing because it was unable to detect improper installation of the differential pressure transmitters and did not demonstrate that the transmitters would function properly with respect to water level in the accumulator.

Actions Taken to Prevent Recurrence

Licensee - A new installation procedure has been issued which now requires verification of proper tubing connections for differential pressure transmitters, and the licensee has committed to strengthening the post modification testing program. Additionally, the licensee reviewed other safety-related differential pressure applications for similar problems. No other problems were identified. Prior to startup of Unit 1, the UHI differential pressure transmitters were properly connected and the system functionally tested using an adequate method. Unit 2 was inspected and found to have the UHI differential pressure transmitters properly connected.

NRC - An inspector from NRC Region II was sent to the site on November 2, 1984, to participate in the investigation of the event. All plants which have UHI systems were determined to be located within NRC Region II, and were notified of this problem. Each licensee reported that the configuration of UHI differential pressure transmitters had been inspected and confirmed to be correct. Inspection and Enforcement Information Notice No. 85-02 (Ref. 9), was sent on January 11, 1985, to all reactor facilities with operating licenses or construction permits to alert them of possible problems associated with improper installation of differential pressure transmitters and inadequate post-modification functional testing.

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Editor's Note

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April 25, 1984 modification described above because automatic closure of the accumulator isolation valves could no longer occur.

The licensee checked Unit 2 and found that the set points also were erroneously set (since February 1983). The cause of the incorrect set points for both Units was due to an engineering error in the calibration procedure which established the set points. The licensee set the set points correctly and revised the appropriate procedures to prevent recurrence.

This violation of plant technical specifications was included in the NRC enforcement action described above (Ref. 10).

On March 22, 1985, the NRC forwarded Inspection and Enforcement Information Notice No. 85-23 (Ref. 11) to all nuclear power reactor facilities holding an operating license or a construction permit to alert them of various problems experienced at the McGuire nuclear power facility in regard to inadequate surveillance and post-maintenance and post-modification system testing.

Unless new significant information becomes available, this incident is considered closed for purposes of this report.

FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the fourth calendar quarter of 1984. As of the date of this report, the NRC had determined that the following was an abnormal occurrence.

84-19 Buildup of Uranium in a Ventilation System

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 10 of "For All Licensees") of this report notes that a major deficiency in design, construction or operation having safety implications requiring immediate remedial action can be considered an abnormal occurrence.

Date and Place - On October 5, 1984, Nuclear Fuel Services, Inc. (the licensee) notified the NRC that an excessive buildup of uranium had been discovered in the new ventilation system (including a scrubber) of the scrap recovery facility at their plant located near the town of Erwin, Tennessee.

Nature and Probable Consequences - Operation of the new ventilation system for the scrap recovery facility began in March 1983. This system was designed to reduce the level of radioactive effluents and projected offsite doses. Within three months of startup, however, the licensee noted higher than expected levels of uranium-235 being accumulated in the ventilation system venturi scrubber and established action limits for the accumulation. In July 1983, a heat exchanger was removed which was later found to contain several hundred grams of uranium-235.

In May 1984, action limits were established in the license which required investigation and corrective actions when the action limits were exceeded. Between May 1984 and October 1984, these action limits were exceeded several times.

On October 3 and 4, 1984, the licensee again detected uranium concentrations in the venturi scrubber solution, which exceeded the license action limit for the liquid. On October 4, 1984, nondestructive assay (NDA) measurements of the venturi scrubber and its blowdown tank showed a buildup of solids containing uranium-235 exceeding the 50-gram action limit. Repeated flushing of the system with water did not reduce the concentration below the action limit. Consequently, the system was shut down, the solution was drained from the scrubber, and the inspection port cover plate was removed for a visual inspection of the scrubber internals.

A buildup of solids on the inner walls of the scrubber and the venturi above the water level was observed. Also, solids were observed to have accumulated in the bottom elbow of the duct where the air enters the scrubber. NDA measurements of the system revealed approximately 1000 grams of uranium-235 in the venturi scrubber and 1000 grams of uranium-235 in the duct leading to the scrubber.

Cleaning of the ventilation system was conducted on October 6 and 7, 1984, after preparation of a procedure and further discussions with the NRC. The NRC resident inspector observed and monitored the licensee's activities.

After reassembly of the scrubber, the removed materials, which had been placed in safe geometry bottles, were measured as containing 1610 grams of uranium-235. An additional 598 grams of uranium-235 were removed from the scrubber in the solution batches of October 3, 4, and 5, 1984. In conjunction with the restarting of the ventilation system, an investigation was initiated by the licensee to determine the causes of the accumulation of uranium-235, and a confirmatory evaluation of the health and safety significance of the observed accumulation was performed.

Even though it was determined that a criticality event could not have occurred, the event was significant in that the accumulation of uranium-235, in the scrubber and ducting, was considerably greater than one safe wet mass.

Additional concerns identified were, (1) the special nuclear material was not maintained within a material balance area as required by license conditions, (2) the special nuclear material was not measured during physical inventories as required by license conditions, and (3) the special nuclear material in the duct work was not stored as specified by the Physical Security Plan. (Although the material was not stored as required, it was not vulnerable.)

<u>Cause or Causes</u> - The primary cause of the uranium buildup was equipment design. The licensee had attempted to design the ventilation system so that all solid materials entering the ventilation ducting would be carried through and into the scrubber where it would be routinely removed. However, the existence of acid and moisture in the air caused the solid material to deposit in the ducting and above the waterline in the scrubber. A contributing cause was the licensee's failure to take appropriate corrective actions when action limits were exceeded.

Preliminary findings from the licensee's investigation indicated that the material removed from the system was from essentially all processes in the scrap recovery operation. The most significant sources were the scrap furnace and the scrap dissolvers. Also, the licensee determined that HEPA filters on other process equipment may have leaked, and a potential existed for liquid to enter the ventilation ducting because of inadequate siphon breaks.

The warning signals, which included the uranium concentrations in the scrubber water and some detected presence of solids during the period May 1984 to October 1984, as well as the accumulation of material in the heat exchanger, were not recognized by the licensee. In two cases when the license action limits were exceeded, no investigation was performed by the licensee. In the other cases, the licensee's investigation consisted of a form filled out by the production foreman with inadequate followup by site management to determine the cause of the condition; in addition, no development of corrective actions to prevent recurrence was made.

Actions Taken to Prevent Recurrence

Licensee - The licensee implemented a program for routinely monitoring material accumulation in the ventilation systems throughout the plant. The NRC approved a license amendment incorporating this monitoring program. The licensee is conducting a design review to identify engineering improvements which will prevent uranium from entering the ventilation system. Status reports on these engineering improvements are provided to the NRC on a routine basis. In addition, the licensee will respond to the NRC enforcement action described below.

NRC - A special inspection was conducted at the licensee's Erwin, Tennessee facility by the NRC Region II Office during the period of October 5-18, 1984. Significant failures to comply with NRC regulatory requirements were identified, i.e., failure to perform adequate investigations and take appropriate corrective actions, as required by the license, for violations of criticality safety action limits placed on the accumulation of uranium in the ventilation system. The conditions of degraded safety and safeguards had existed for a significant period of time.

An Enforcement Conference to discuss these matters was held with the licensee at the NRC Region II Office on October 29, 1984. On February 21, 1985, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $20,000 (Ref. 12) In addition to the civil penalty, the NRC believed that further remedial action was needed to ensure that the licensee improves management oversight of operations and initiates appropriate investigations when action limits are exceeded. Therefore, the February 21, 1985, NRC letter (Ref. 12) also enclosed an Order Modifying License. The Order amends the license to require the licensee to expand the duties and responsibilities of its Internally Authorized Change Council.

The NRC Resident Inspector is monitoring the licensee's on-site actions. Both the Resident Inspector and the NRC Region II Office are following the licensee's corrective actions to assure that they are satisfactory.

Unless new significant information becomes available, this incident is considered closed for purposes of this report.

OTHER NRC LICENSEES

(Industrial Radiographers, Medical Institutions, Industrial Users, etc.)

There are currently more than 8,000 NRC nuclear material licenses in effect in the United States, principally for use of radioisotopes in the medical, industrial and academic fields. Incidents were reported in this category from licensees such as radiographers, medical institutions, and byproduct material users.

The NRC is reviewing events reported by these licensees during the fourth calendar quarter of 1984. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the fourth calendar quarter of 1984, an Agreement State reported the following abnormal occurrence to the NRC.

AS84-3 Overexposure of a Radiographer Trainee

Appendix A (see Example 1 of "For All Licensees") of this report notes that exposure of the feet, ankles, hands, or forearms of any individual to 375 rems or more of radiation can be considered an abnormal occurrence.

Date and Place - On October 31, 1984, a radiographer trainee, employed by Ultrasonics Specialists, Inc., (a Louisiana industrial radiography licensee), of Amelia, Louisiana, received a significant exposure to his right index finger, while working at Avondale Shipyards in Morgan City, Louisiana.

Nature and Probable Consequences - The trainee attempted to disconnect a source tube from a Gulf Nuclear Model 40V exposure device containing approximately 43 curies of iridium-192. A radiation survey meter was not used during the disconnecting operation. After disconnecting the source tube, the trainee noted that the drive cable could be seen and that the source was still in the exposed position. Based on discussions with the trainee, he believed his right index finger may have slid along the cable and may have come into contact with the source.

About four days after the incident, he experienced pain in the finger. About 15 days after the incident, the finger developed a blister. The individual is receiving medical treatment for the injury.

During the incident, the individual's pocket dosimeter was discharged beyond its range; however, his TLD (thermoluminescent dosimeter), when processed by the badge supplier, only showed a whole body dose of 163 millirem. A reenactment of the incident was made and it was estimated that the dose to the finger was between 2,500 and 3,500 rads.

Cause or Causes - The two principal factors contributing to the overexposure were (1) the trainee was performing industrial radiography without the direct supervision of an authorized instructor, and (2) a survey meter, which would have indicated that the source was not in the shielded position, was not used.

Actions Taken to Prevent Recurrence

Licensee - Appropriate corrective actions will be required in response to violations cited by the State Agency.

Louisiana Nuclear Energy Division - The State Agency investigated the event. Appropriate violations were cited for the excessive exposure and for the radiographer trainee performing radiography without direct supervision by an authorized instructor. Additionally, because the incident resulted in an injury, a penalty of $5,000 was assessed.

This incident is closed for purposes of this report.

REFERENCES

1. Confirmatory Action Letter No. 84-18 from Richard W. Starostecki, Director, Division of Project and Resident Programs, NRC Region I, to Bruce Kenyon, Vice President - Nuclear Operations, Pennsylvania Power & Light Company, Docket Nos. 50-387/50-388, October 17, 1984.*

2. Letter from Bruce D. Kenyon, Vice President - Nuclear Operations, Pennsylvania Power & Light Company, to Richard W. Starostecki, Director, Division of Project and Resident Programs, NRC Region I, Docket Nos. 50-387/50-388, November 19, 1984.*

3. Letter from Bruce D. Kenyon, Vice President - Nuclear Operations, Pennsylvania Power & Light Company, to Richard W. Starostecki, Director, Division of Project and Resident Programs, NRC Region I, Docket Nos. 50-387/50-388, January 9, 1985.*

4. Letter from Richard W. Starostecki, Director, Division of Project and Resident Programs, NRC Region I, to Bruce D. Kenyon, Vice President - Nuclear Operations, Pennsylvania Power & Light Company, Docket Nos. 50-387/50-388, November 15, 1984.*

5. Letter from Richard W. Starostecki, Director, Division of Project and Resident Programs, NRC Region I, to Bruce D. Kenyon, Vice President - Nuclear Operations, Pennsylvania Power & Light Company, Docket Nos. 50-387/50-388, January 10, 1985.*

6. Letter from Richard W. Starostecki, Director, Division of Project and Resident Programs, NRC Region I, to Bruce D. Kenyon, Vice President - Nuclear Operations, Pennsylvania Power & Light Company, forwarding a Notice of Violation, Docket No. 50-387, January 4, 1985.*

7. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-27, "Notification to the NRC Operations Center and Reporting Events on Licensee Event Reports," April 3, 1985.*

8. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 84-45, "Reversed Differential Pressure Instrument Sensing Lines," June 11, 1984.*

9. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-02, "Improper Installation and Testing of Differential Pressure Transmitters," January 11, 1985.*

10. Letter from J. Nelson Grace, Regional Administrator, NRC Region II, to H.B. Tucker, Vice President, Nuclear Production Department, Duke Power Company, forwarding (a) Notice of Violation and Proposed Imposition of Civil Penalty, and (b) Inspection Report Nos. 50-369/84-34 and 50-370/84-31, Docket Nos. 50-369 and 50-370, February 20, 1985*.

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*Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555 for inspection and copying (for a fee).

11. U. S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-23," Inadequate Surveillance and Postmaintenance and Postmodification System Testing," March 22, 1985.*

12. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to C.W. Taylor, President, Nuclear Fuel Services, Inc., forwarding (a) Notice of Violation and Proposed Imposition of Civil Penalty, and (b) Order Modifying License, Docket No. 70-143, February 21, 1985*.

*Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555 for inspection and copying (for a fee).

APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

The following criteria for this report's abnormal occurrence determinations were set forth in an NRC policy statement published in the Federal Register on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952).

Events involving a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to:

1. Moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission;

2. Major degradation of essential safety-related equipment; or

3. Major deficiencies in design, construction, use of, or management controls for licensed facilities or material.

Examples of the types of events that are evaluated in detail using these criteria are:

For All Licensees

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any idnvidual to 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation (10 CFR §20.403(a)(1)), or equivalent exposures from internal sources.

2. An exposure to an individual in an unrestricted area such that the whole-body dose received exceeds 0.5 rem in one calendar year (10 CFR §20.105(a)).

3. The release of radioactive material to an unrestricted area in concentrations which, if averaged over a period of 24 hours, exceed 500 times the regulatory limit of Appendix B, Table II, 10 CFR Part 20 (10 CFR §20.403(b)).

4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as (a) a radiation dose rate of 1,000 mrem per hour three feet from the surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit.

5. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.

6. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.

7. Any substantiated loss of special nuclear material or any substantiated inventory discrepancy which is judged to be significant relative to normally expected performance and which is judged to be caused by theft or diversion or by substantial breakdown of the accountability system.

8. Any substantial breakdown of physical security or material control (i.e., access control, containment, or accountability systems) that significantly weakened the protection against theft, diversion or sabotage.

9. An accidental criticality (10 CFR §70.52(a)).

10. A major deficiency in design, construction or operation having safety implications requiring immediate remedial action.

11. Serious deficiency in management or procedural controls in major areas.

12. Series of events (where individual events are not of major importance), recurring incidents, and incidents with implications for similar facilities (generic incidents), which create major safety concern.

For Commercial Nuclear Power Plants

1. Exceeding a safety limit of license technical specifications (10 CFR §50.36(c)).

2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

3. Loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

4. Discovery of a major condition not specifically considered in the safety analysis report (SAR) or technical specifications that requires immediate remedial action.

5. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

For Fuel Cycle Licenses

1. A safety limit of license technical specifications is exceeded and a plant shutdown is required (10 CFR §50.36(c)).

2. A major condition not specifically considered in the safety analysis report or technical specifications that requires immediate remedial action.

3. An event which seriously compromised the ability of a confinement system to perform its designated function.

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the October through December, 1984 period, the NRC, NRC licensees, Agreement States, Agreement States Licensees, and other involved parties, such as reactor vendors and architects and engineers, continued with the implementation of actions necessary to prevent recurrence of previously reported abnormal occurrences. The referenced Congressional abnormal occurrence reports below provide the initial and any updating information on the abnormal occurrences discussed. Those occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

79-3 Nuclear Accident at Three Mile Island

This abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and updated in each subsequent report in this series, i.e., NUREG-0090, Vol. 2, No. 2 through Vol. 7, No. 3. It is further updated as follows:

Reactor Building Entries

During the fourth calendar quarter of 1984, 58 entries were made into containment. There have been a total of 522 entries since the March 28, 1979 accident. Reactor building activities during the fourth quarter included polar crane inspections, the jacking of the reactor vessel upper plenum assembly, and the inspection of the basement of the reactor building by a robotic vehicle. The latter two items are discussed below.

Plenum Jacking

During December 1984, four hydraulic jacks, positioned inside the reactor vessel, were used to raise the plenum assembly to 7¼ inches above its normal seating surface. It was anticipated that the jacks would be used to raise the plenum to 9 inches; however, structural interferences between the jacks and the plenum limited the lift to 7¼ inches. The plenum was lifted without any indication of binding and no obstacles were noted which may interfere with plenum removal. The polar crane will be used to complete the plenum lift and to transfer the plenum to storage in the deep end of the refueling canal.

During the plenum jacking, long handled tools were used to detach the remaining fuel assemblies and fuel assembly end fittings which adhered to the underside of the plenum. The plenum will remain supported on jack mechanical followers until its scheduled removal in May 1985.

Rcbotic Inspection of Reactor Building Basement

During November 1984, a robotic vehicle was used to inspect the basement of the reactor building. The inspections were limited to the areas outside of the "D" rings. An expanded program for the robotic vehicles is currently being evaluated. Dose rates measured on the 282 ft. elevation during the inspection confirmed previous measurements that the basement is essentially inaccessible to humans. General area radiation fields range from 10 to 70 R/hr. One location near the elevator enclosure was measured to be in excess of 1,100 R/hr.

TMI Annual Emergency Exercise

The TMI site annual emergency exercise was held on October 3, 1984. Licensee actions in responding to a GPU Nuclear developed reactor emergency scenario were evaluated by an NRC Region I inspection team. Among the activities evaluated were: detection, classification and operational assessment of the simulated emergency; post-accident sampling systems; emergency damage analysis, control and repair; direction and coordination of the licensee's emergency response; radiological monitoring capabilities in-plant, onsite and offsite; notifications of licensee personnel and offsite agencies; plant and site security; radiological consequence analyses for radioactive releases in-plant, onsite and to the public offsite; and performance of first aid and rescue. Engineering responses from onsite and remotely located engineering groups were evaluated. Communications information flow and record keeping, in-plant and between the licensee's emergency support centers and local, state and federal agencies were also evaluated. The NRC inspection team reviewing the exercise identified no violations, and determined that within the confines of the scenario, actions taken were adequate to protect the health and safety of the public.

EPICOR-II/Submerged Demineralizer System (SDS) Processing

The EPICOR-II System processed approximately 87,435 gallons of water during the fourth quarter of 1984. The SDS processed approximately 83,724 gallons of water during the same time period.

Liner Shipments

Three shipments of spent resins were sent from the TMI site to Hanford, Washington, during this reporting period. One shipment was sent from TMI to the U.S. Ecology Waste Management Facility at Richland, Washington.

TMI 2 Commission Meeting

On November 7, 1984, the NRC Commissioners met on the status of cleanup activities and funding for Three Mile Island Unit 2. Separate presentations were made by both the NRC staff and representatives of GPU Nuclear. Edwin E. Kintner, Executive Vice President for GPU Nuclear, provided a summary of the progress made thus far in the cleanup and indicated that removal of damaged fuel from the reactor is scheduled to begin in July 1985. The NRC staff noted its optimism regarding committed funding for the cleanup. While pointing out that there are many uncertainties associated with securing large funds from diverse sources, the NRC staff stated that there is reasonable assurance that the licensee will be successful in securing the needed cleanup funding.

TMI-2 Advisory Panel Meetings

The Advisory Panel for the Decontamination of Three Mile Island Unit 2 (Panel) met on October 11, 1984 in Harrisburg, Pennsylvania. At this meeting the Panel reported on issues relative to the TMI-2 cleanup effort contained in specific TMI-1 restart NRC Commission Meeting transcripts. The licensee presented a briefing on the proposed removal of the reactor pressure vessel plenum and other activities to be accomplished prior to commencement of fuel removal.

The Panel also met on November 8, 1984 in Lancaster, Pennsylvania. At this meeting the Panel received reports from the licensee and U.S. Environmental Protection Agency on the results of onsite and offsite krypton-85 monitoring during the reactor pressure vessel head lift operation. The Department of Health, Commonwealth of Pennsylvania, presented a summary of health related studies conducted in the vicinity of the TMI site since the accident. The licensee also briefed the Panel on their investigation regarding misadjustment of the brakes on the reactor building polar crane.

The Panel's newest member, Joseph J. DiNunno, attended this meeting. Mr. DiNunno, an independent consultant, holds B.S. and M.S. degrees in Electrical Engineering from Pennsylvania State University and the University of Maryland, respectively. He is a resident of Annapolis, Maryland, and replaces Dr. Henry Wagner, also a Maryland resident, who has resigned from the Panel.

Future reports will be made as appropriate.

* * * * * *

84-2 Through Wall Crack in Vent Header Inside BWR Containment Torus

This abnormal occurrence was originally reported in NUREG-0090 Vol. 7, No. 1, "Report to Congress of Abnormal Occurrences: January-March 1984." It is updated as follows.

As discussed in the previous report, on February 3, 1984, a through wall crack was discovered in the vent header within the containment torus which degraded the containment pressure suppression capability of Georgia Power Company's Hatch Unit 2, a boiling water reactor (BWR) located in Appling County, Georgia. The event raised a possible generic concern for other BWR plants which utilize similar containment and inerting system designs.

Inspection and Enforcement Bulletin No. 84-01 (Ref. B-1), dated February 3, 1984, was issued to all BWRs with operating licenses or construction permits. The bulletin requested that facilities with operating licenses in cold shutdown and with primary containments similar to the Hatch containment (Mark I) perform inspections as to the condition of their vent headers and report the results to NRC. All applicable licensees made the required inspections; no cracking or relevant indications were found. The NRC closed out the bulletin.

On March 5, 1984, IE Information Notice No. 84-17 (Ref. B-2) was sent to all reactor facilities with operating licenses or construction permits to alert them to possible problems associated with cooling components to below their nil ductility temperatures with liquid nitrogen. The Notice also advised licensees and applicants of potentially similar problems associated with the use of other

very cold fluids where the fluid could come in contact with safety-related components subject to brittle fracture.

In response to the Hatch 2 event, a BWR Regulatory Response Group (RRG) was activated. This group, together with the vendor (GE) representatives, met with the NRC on February 6 and 23, 1984, to discuss actions to be taken to assure the integrity of the containment and associated systems. The agreed upon recommendations were documented in GE Service Information Letter (SIL) No. 402 (Ref. B-3). As stated in the previous report, these actions involve evaluations of inerting system design and operation, performance of a leakage test to confirm the integrity of the vent system, inspection of the nitrogen injection line, and inspection of containment components and equipment. The owners' group letter transmitting the SIL requested that the licensees and applicants report their findings to the NRC.

All responses were submitted to the NRC by September 1984. The NRC staff has reviewed the responses to the SIL recommendations and has completed its report to close out this issue (Ref. B-4).

This incident is closed for purposes of this report.

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84-9 Degraded Shutdown Systems

This abnormal occurrence was originally reported in NUREG-0090, Vol. 7, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1984." It is updated as follows.

Presently, the Fort St. Vrain (FSV) reactor remains shutdown while the licensee, Public Service Company of Colorado (PSC), continues investigations and actions necessary to correct the problems described in the previous report. Also, as discussed in the previous report, the NRC issued a preliminary report (Ref. B-5) on October 16, 1984, which assessed FSV operations and several problem areas at the plant.

In subsequent meetings between NRC and PSC staffs, the status of various licensing and inspection issues including the NRC assessment report findings and PSC commitments for implementing NRC recommendaions have been discussed. As stated in an NRC letter to the licensee, dated January 18, 1985 (Ref. B-6), PSC has committed to perform various actions prior to the projected restart date of April 1, 1985. The major actions are as follows.

1. Inspect and refurbish all control rod drive and orifice assemblies. the refurbishment will include the replacement of the old 347 stainless steel suspension cables with new Inconel 625 cables, the replacement of the old boronated graphite balls with new material having a lower B_2O_3 content, and the cleaning and/or replacement of bearings and other components as necessary.

2. Utilize back-electromagnetic force (EMF) testing to obtain scram signature traces of all control rod drive mechanisms in order to enable the development of in pile acceptance criteria.

3. Establish procedures for conducting periodic control rod drop tests.

4. Upgrade position and temperature instrumentation on all control rod drive and orifice assemblies.

5. Provide the results of an independent (third party) review of the PSC management structure and practices relative to the operation of FSV.

6. Propose various revisions and upgrades to the FSV Technical Specifications.

7. Develop a surveillance program that will demonstrate continued acceptability of the prestressed concrete reactor vessel tendon system.

The PSC has documented and submitted for NRC review the details of these short-term actions and provided commitments and schedules for some long-term actions, in order to provide sufficient time for NRC review prior to restart.

Future reports will be made as appropriate.

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84-15 <u>Significant Internal Exposure to Iodine-125</u>

This abnormal occurrence was originally reported in NUREG-0090, Vol. 7, No. 3, "Report to Congress on Abnormal Occurences: July-September 1984." It is updated as follows.

The event involved a laboratory employee of the Veterans Administration Medical Center, Bronx, New York, who was found to have a thyroid burden of about 524 microcuries of iodine-125. The licensee estimated a total absorbed dose to the individual's thyroid of about 2000 rads.

On March 5, 1985, the NRC forwarded to the licensee a Notice of Violation and and Order Modifying License (Ref. B-7). The licensee was cited for alleged failure to protect a laboratory worker from exposure to radioactive material in an amount in excess of regulatory limits. A second violation was cited for failure of an individual to wear disposable gloves while working with licensed material.

To emphasize the importance of adherence to NRC requirements and safe performance of licensed activities, an Order Modifying License was issued to require periodic unannounced audits of the licensee's radiation safety program by an independent third party. The order requires the third party to observe the actions of the licensee's employees while involved in the use of licensed material to verify adherence to NRC requirements.

Unless new significant information becomes available, this incident is considered closed for purposes of this report.

APPENDIX C

OTHER EVENTS OF INTEREST

The following item is described below because it may possibly be perceived by the public to be of public health significance. The item did not involve a major reduction in the level of protection provided for public health or safety; therefore, it is not reportable as an abnormal occurrence.

1. Contamination of Sanitary Sewage Systems

Three recent occurrences have pointed out the need to focus attention on the possible reconcentration of radionuclides that are discharged into sanitary sewage systems under the provisions of 10 CFR §20.303. In these occurrences, radioactive contamination was found in sewer lines and in municipal sewage treatment facilities. While any exposures to people would be small and considered medically insignificant, the events represented unnecessary hazards to treatment facility personnel. In addition, unnecessary expense and efforts were required to determine the extent of the problems and the proper disposition of the contaminated material.

NRC regulation 10 CFR §20.303 permits discharges of small quantities of radionuclides into sanitary sewage systems, within the limits specified in that section, provided that the materials "are readily soluble or dispersible in water." The purpose of these requirements is to assure that when the radioactive material is mixed with the very large amounts of fluid in the municipal sewer systems, the radioactivity per milliliter (ml) would be virtually negligible. This would assure that the sludge or ash (if incineration is used as a final step) produced by sewage treatment facilities would also contain virtually negligible amounts of radioactivity per unit weight.

Licensees who rely on 10 CFR §20.303 have the burden of demonstrating that the materials they are discharging are indeed readily soluble or dispersible. There have been cases where the term "dispersible" may have been misapplied, leading to licensees introducing substances into sanitary sewage systems that do not qualify as readily dispersible, such as liquid scintillation media and ash (some licensees may produce ash from incinerating radioactively contaminated rags, clothing, filters, etc., which may then be placed into sanitary sewage systems.) Ash is a special case, which may or may not be "readily dispersible," depending on its degree of comminution and tendency to agglomerate.

It should be noted from the events described below that sewage treatment processes and disposal methods for sewage products such as sludge or incineration products may result in concentration of radioactive materials disposed into sanitary sewers, thereby creating unnecessary hazards and concerns.

During a routine radiation survey, Oak Ridge Associated Universities found radioactive contamination in the sludge from the sewage treatment facility in Oak Ridge, Tennessee. The principal contaminant was cobalt-60 (Co-60). The State of Tennessee traced the apparent source of the contamination to a State

licensee who occasionally discharged a few thousand gallons per day of liquid into the sanitary system at concentrations of 66-110 dpm (disintegrations per minute)/ml of Co-60. Although the discharge from the licensee's facility was mixed with 4 to 5 million gallons of liquid from other sources in the city, concentrations of 20,000-200,000 dpm/Kg were measured in the sludge from the treatment facility. Sludge had been used to fertilize a Department of Energy reforested area with the result that radiation levels two to three times background were measured there (about 10 μR/hr). As a result of the discovery of the problem, the licensee has installed an improved filtration and ion exchange system.

In the second occurrence, americium-241 (Am-241) contamination was found in ash that remained in an incinerator used as a final treatment step at the Tonawanda, New York sewage treatment plant and in ash disposed of at the Tonawanda landfill. About 10,000 tons of ash containing about 500 picocuries per gram of ash have been disposed of at the landfill. About 30 tons of contaminated ash currently remain in the sewage treatment plant incinerator and in ancillary equipment.

The contamination resulted from liquid releases made to the sanitary sewage system by a New York Agreement State licensee who formerly manufactured Am-241 foils at its Tonawanda facility. The licensee has since relocated its foil manufacturing operation to Mexico. (The NRC Office of International Programs has been in contact with their counterpart in Mexico regarding this licensee.) Some decontamination of the licensee's Tonawanda facility was undertaken following the move.

The homes, clothes, or cars of four former licensee employees were found to be contaminated. Because of exposure to airborne dust at the sewage treatment plant, six plant workers received whole-body counting to examine the potential for internal deposition. Two of the six were among 58 plant workers who received lung scans. No uptake of Am-241 was detected. Several issues remain to be resolved. These involve disposal of the contaminated ash at the sewage treatment plant, dealing with contaminated ash disposed to the landfill before identification of the contamination problem, decontamination of the sewer lines, and decontamination of the licensee's facility. The State of New York is presently negotiating with a contractor to begin the decontamination process. The State has requested the contractor to perform an evaluation of the solubility and dispersibility characteristics of the americium involved in this incident.

In a third occurrence, Am-241 contamination also was found in sludge at a sewage treatment plant in Grand Island, New York. The contamination resulted from liquid releases made to the sanitary sewage system by another New York Agreement State licensee also engaged in the manufacture of Am-241 foils. The measured concentration in the sludge was about 100 picocuries per gram of sludge dry weight. (For purposes of comparison, if this sludge were incinerated, concentrations of about 500 picocuries per gram would result in the ash produced.) In this case, however, the sludge is disposed directly to a local sanitary landfill.

As previously mentioned, the amounts of radioactivity at the treatment plants were quite small for all three events, therefore the impact on public health or safety was insignificant. On this basis, the events are not reportable as an abnormal occurrence. They are being reported here because of their general

interest. In addition, some of the events received considerable media coverage in the general areas of the States involved.

The information described above was forwarded on December 21, 1984, to all NRC materials licensees other than licensees that use sealed sources only, by Inspection and Enforcement Information Notice No. 84-94 (Ref. C-1).

REFERENCES
(FOR APPENDICES)

B-1 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 84-01, "Cracks in Boiling Water Reactor Mark I Containment Vent Headers," February 3, 1984.*

B-2 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 84-17, "Problems with Liquid Nitrogen Cooling Components Below the Nil Ductility Temperature," March 5, 1984.*

B-3 General Electric Nuclear Services Operations, Service Information Letter (SIL) No. 402, "Wetwell/Drywell Inerting," February 14, 1984.*

B-4 Letter from D.G. Eisenhut, Director, Division of Licensing, NRC Office of Nuclear Reactor Regulation, to E.L. Jordan, Director, Division of Emergency Preparedness and Engineering, NRC Office of Inspection and Enforcement, "BWR Vent Header Cracking," February 12, 1985.*

B-5 Letter from Harold R. Denton, Director, NRC Office of Nuclear Reactor Regulation, to R. F. Walker, President, Public Service Company of Colorado, forwarding "Preliminary Report Related to the Restart and Continued Operation of Fort St. Vrain Nuclear Generating Station," Docket No. 50-267, October 16, 1984.*

B-6 Letter from R. D. Martin, Regional Administrator, NRC Region IV, to O. R. Lee, Vice President, Public Service Company of Colorado, Docket No. 50-267, January 18, 1985.*

B-7 Letter from J. M. Taylor, Director, NRC Office of Inspection and Enforcement, to K. L. Mulholland, Jr., Director, Veterans Administration Medical Center, Bronx, New York, forwarding an Order Modifying License and a Notice of Violation, Docket No. 30-17020, March 5, 1985.*

C-1 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 84-94, "Reconcentration of Radionuclides Involving Discharges Into Sanitary Sewage Systems Permitted Under 10 CFR §20.303," December 21, 1984.*

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*Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555 for inspection and copying (for a fee).

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** SEE INSTRUCTIONS ON THE REVERSE	1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any) NUREG-0090 Vol. 7, No. 4
2. TITLE AND SUBTITLE Report to Congress on Abnormal Occurrences October-December 1984	3. LEAVE BLANK
	4. DATE REPORT COMPLETED MONTH May YEAR 1985
5. AUTHOR(S)	6. DATE REPORT ISSUED MONTH May YEAR 1985
7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Office for Analysis and Evaluation of Operational Data U.S. Nuclear Regulatory Commission Washington, DC 20555	8. PROJECT/TASK/WORK UNIT NUMBER
	9. FIN OR GRANT NUMBER
10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Office of Analysis and Evaluation of Operational Data U.S. Nuclear Regulatory Commission Washington, DC 20555	11a. TYPE OF REPORT Quarterly b. PERIOD COVERED (Inclusive dates) October-December 1984
12. SUPPLEMENTARY NOTES	

13. ABSTRACT (200 words or less)

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 to December 31, 1984. During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved four control rods failing to insert during testing and the other involved degraded upper head injection system accumulator isolation valves. There was one abnormal occurrence at a fuel cycle facility; the event involved buildup of uranium in a ventilation system. There was one abnormal occurrence reported by an Agreement State; the event involved an overexposure of a radiographer trainee. The report also contains information updating some previously reported abnormal occurrences.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS	15. AVAILABILITY STATEMENT Unlimited
	16. SECURITY CLASSIFICATION (This page) Unclassified (This report) Unclassified
b. IDENTIFIERS/OPEN ENDED TERMS Control Rods; Control Rod Testing; Scram Pilot Solenoid Valves; Instrument Air; Contaminated Valve Seat; Upper Head Injection System; Isolation Valves; Differential Pressure Transmitters; Uranium Buildup; Ventilation Systems; Overexposure to Radiographer Trainee.	17. NUMBER OF PAGES
	18. PRICE

*U.S. GOVERNMENT PRINTING OFFICE: 1985-461-721:20122

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

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