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

CHAIRMAN

February 18, 1987

The Honorable George H. W. Bush
President of the Senate
Washington, DC 20510

Dear Mr. President:

Enclosed is the NRC report on abnormal occurrences at licensed nuclear facilities, as required by Section 208 of the Energy Reorganization Act of 1974 (PL 93-438), for the second calendar quarter of 1986.

In the context of the Act, an abnormal occurrence is an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. The report states that for this reporting period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved an out of sequence control rod withdrawal and the other involved a boiling water reactor emergency core cooling system design deficiency. There were five abnormal occurrences at the other NRC licensees. Two involved willful failure to report diagnostic medical misadministrations to the NRC, one involved a therapeutic medical misadministration and two involved diagnostic medical misadministrations. There were two abnormal occurrences reported by the Agreement States. One involved an uncontrolled release of krypton-85 to an unrestricted area. The other involved a contaminated radiopharmaceutical used in diagnostic administrations.

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

In addition to this report, we will continue to disseminate information on reportable events. These event reports are routinely distributed on a timely basis to the Congress, industry, and the general public.

Sincerely,

Lando W. Zech, Jr.

Lando W. Zech, Jr.
Chairman

Enclosure:
Report to Congress on
Abnormal Occurrences
(NUREG-0090, Vol. 9, No. 2)

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CHAIRMAN

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

February 18, 1987

The Honorable Jim Wright
Speaker of the United States
House of Representatives
Washington, DC 20515

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April - June 1986

U.S. Nuclear Regulatory Commission

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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 April 1 to June 30, 1986.

The report states that for this reporting period, there were two abnormal occurrence at the nuclear power plants licensed to operate. One involved an out of sequence control rod withdrawal and the other involved a boiling water reactor emergency core cooling system design deficiency. There were five abnormal occurrences at the other NRC licensees. Two involved willful failure to report diagnostic medical misadministrations to the NRC; one involved a therapeutic medical misadministration; and two involved diagnostic medical misadministrations. There were two abnormal occurrences reported by the Agreement States. One involved an uncontrolled release of krypton-85 to an unrestricted area; the other involved a contaminated radiopharmaceutical used in diagnostic administrations.

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 April 1 to June 30, 1986.

Information reported on each events 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 method 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 Supplements 1 and 2 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 July 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 in 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.

In early 1977, the Commission determined that abnormal occurrences happening at facilities of Agreement State licensees should be included in the quarterly reports to Congress. The abnormal occurrence criteria included in Appendix A is 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
APRIL-JUNE 1986

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the second calendar quarter of 1986. As of the date of this report, the NRC had determined that the following event was an abnormal occurrence.

86-8 Out of Sequence Control Rod Withdrawal

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 11 of "For All Licensees") notes that serious deficiency in management or procedural control in major areas can be considered an abnormal occurrence.

Date and Place - On March 18, 1986, during a startup of Peach Bottom Unit 3, personnel errors by four licensed operators resulted in a control rod being withdrawn out-of-sequence without being detected by these operators. The next operating shift detected the error and manually scrammed the unit. Peach Bottom Unit 3 is a General Electric-designed boiling water reactor (BWR) operated by Philadelphia Electric Company (the licensee) and located in York County, Pennsylvania.

Nature and Probable Consequences - To limit reactivity additions during startup and low power operations, BWRs utilize a variety of procedural controls and monitoring systems. The reactor operator follows a control rod withdrawal sequence procedure developed by reactor engineers. A rod worth minimizer (RWM) computer system monitors this sequence and blocks control rod movement upon significant deviation from the prescribed sequence. A rod sequence control system (RSCS) complements and backs up the RWM in restricting control rod movement. A major purpose of these controls is to limit the consequences of a control rod drop event during the startup.

On March 17, 1986, a startup was begun on Peach Bottom Unit 3. The RWM was bypassed due to a computer hardware fault. As allowed by technical specifications, a second licensed reactor operator was assigned to monitor another licensed reactor operator as the latter withdrew control rods in accordance with the sequence prescribed by procedure. At 1:28 a.m. on March 18, 1986, the operator withdrew control rod 10-23 out of sequence instead of rod 02-23. The second operator monitoring the rod withdrawals failed to notice the error.

Later, at its prescribed place in the procedure, both operators signed off the withdrawal of control rod 10-23. Shortly afterward, the reactor attained criticality. At 2:30 a.m., withdrawal of additional control rods in an attempt to increase reactor power was blocked by the RSCS since rod 02-23 was not in its prescribed position. The Shift Superintendent and Shift Supervisor then bypassed the RSCS rod 02-23 full out logic with a keylock switch without verifying the rod position and conformance to the rod withdrawal sequence as required by the procedure for bypassing RSCS logic.

Rod withdrawal and startup continued with rod 02-23 fully inserted instead of being fully withdrawn as required.

After 7:00 a.m., the oncoming shift requested that the RWM be returned to service. This was accomplished at 7:38 a.m.; the operators noted an insert error for rod 02-23. The rod was confirmed to be out of position for the sequence. The Shift Supervisor returned the RSCS bypass for control rod 02-23 to normal. Two control rods were inserted and then the reactor was manually scrammed from approximately 3% power at 8:55 a.m. The NRC Senior Resident Inspector and Duty Officer were notified of the scram and the out-of-sequence rod shortly afterward.

The licensee presented an analysis of potential consequences of a rod drop event for various rod pairs for the March 18 event. The peak enthalpy deposition in a fuel pin, had a rod drop occurred with rod 02-23 inserted, for the worst case was calculated to be 118 cal/gm. This is less than the peak enthalpy deposition of 215 cal/gm from the reload analysis for the current fuel cycle and the 280 cal/gm design criterion.

Cause or Causes - The out-of-sequence control rod withdrawal resulted from numerous personnel errors by four licensed operators. One licensed reactor operator withdrew the wrong control rod from the core. The RWM, designed to detect such an occurrence, was inoperable. A second licensed operator was assigned to independently verify the correct rod withdrawal sequence as required by Peach Bottom Technical Specifications; he did not identify the error. When the point in the sequence to withdraw the rod already incorrectly withdrawn was reached, neither reactor operator identified the previous error. The Shift Supervisor and Shift Superintendent who were overseeing the startup activities failed to note the error. Further, they bypassed the RSCS without assuring that the bypassed control rod was in its correct position, as required by the procedure for use of the bypass keys. These personnel errors by four licensed individuals showed an inattention to detail and failures to adhere to procedural requirements, possibly resulting from complacent attitudes.

Actions Taken to Prevent Recurrence

Licensee - The four individuals involved in this event were disciplined. Plant staff management meetings were held with all operations personnel to discuss the event and their individual responsibilities. Procedural controls were strengthened to, among other things, use best efforts to place the RWM in service, dedicate a second operator to sequence verification if RWM is bypassed, generate rod position maps at specific withdrawal points and compare with prepared rod position maps and require positive rod position verification prior to RSCS bypass.

NRC - After notification of the out-of-sequence control rod withdrawal by the licensee on March 18, a special safety inspection into the event was conducted at Peach Bottom Atomic Power Station on March 18-21, 1986. The inspection results were forwarded to the licensee in a letter dated March 25, 1986 (Ref. 1).

An enforcement conference was held at NRC Region I on March 27, 1986, between NRC and licensee personnel to discuss the causes of the event and the corrective actions taken and planned. A Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $200,000, dated June 9, 1986, was sent to the licensee (Ref. 2) describing the violations resulting from numerous personnel errors by several licensed personnel.

The enforcement letter stated that these personnel errors indicate that a pattern of inattention to detail, failure to adhere to procedural requirements, and a generally complacent attitude of staff toward performance of their duties continues to exist at Peach Bottom. Since 1983, the licensee has been cited three times, and civil penalties imposed, for violations pertaining to personnel not following procedures. The latest incident demonstrates that the actions taken to correct this pattern have not been effective. Such problems are indicative of a lack of management involvement in and attention to station activities to assure that the station personnel respect, understand the need for, and adhere to licensee policies and procedures for the safe operation of the facility. The proposed civil penalty of $200,000 represented a 100% escalation because: (1) in each case, an opportunity existed for a licensed individual to detect and correct the rod pull error, but the error was not recognized, and (2) the enforcement history at Peach Bottom regarding personnel adherence to procedures has been poor.

This item is considered closed for the purposes of this report.

* * * * * * * * * * *

86-9 <u>Boiling Water Reactor Emergency Core Cooling System Design Deficiency</u>

The following information pertaining to this event is also being reported concurrently in the <u>Federal Register</u>. Appendix A (see the third general criterion) of this report notes that major deficiencies in design, construction, use of, or management controls for licensed facilities or material can be considered an abnormal occurrence. In addition, Examples 10 and 12 of "For All Licensees" of Appendix A are applicable. Example 10 notes that a major deficiency in design having safety implications requiring immediate remedial action can be considered an abnormal occurrence. Example 12 notes that incidents with implications for similar facilities (generic incidents) which create major safety concern can be considered an abnormal occurrence.

<u>Date and Place</u> - On May 19, 1986, the Boston Edison Company (BECO) notified the NRC that a significant design deficiency in the residual heat removal (RHR) system minimum flow protection logic at the Pilgrim Nuclear Power Station (PNPS) had been discovered. Pilgrim is a General Electric (GE)-designed boiling water reactor (BWR) located in Plymouth County, Massachusetts.

Later, it was found that some other GE-designed BWRs also contained the same design deficiency.

<u>Background</u> - The RHR system, which operates at low pressure, functions in different modes to remove reactor decay heat under normal and emergency situations (e.g., loss of coolant accident, LOCA). For normal situations, the RHR system can be operated in the shutdown cooling/head spray mode and the steam condensing mode. For emergency situations, the RHR system functions as part of the emergency core cooling system [at Pilgrim, this system is called the core standby cooling system (CSCS)] operating in the containment spray/cooling mode and the low pressure coolant injection (LPCI) mode.

At Pilgrim, to prevent the RHR pumps from running dead headed, each pair of pumps is provided with a minimum flow bypass capability. The minimum flow bypass consists of an orifice flow bypass which allows a flow of approximately 10 percent of rated flow to bypass the reactor vessel and be returned to the suction source. The minimum flow bypass line for each pair of RHR pumps is connected to a single line and controlled by a single minimum flow bypass valve. The minimum flow bypass valves are normally open. The valves will close upon sensing flow in either of the RHR loops.

Prior to the deficiency found at Pilgrim, the NRC had issued on December 13, 1985, Inspection and Enforcement Information Notice No. 85-94, "Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA" (Ref. 3) to all nuclear power reactor facilities holding an operating license or a construction permit. The Notice alerted licensees of recent instances at various BWR and pressurized water reactor (PWR) plants where it was discovered that minimum flow requirements might not or could not be met for some ECCS pumps under small-break LOCA conditions. The Notice suggested that licensees review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems at their facilities.

Nature and Probable Consequences - While reviewing Information Notice No. 85-94 for applicability to Pilgrim, BECO discovered that a single failure under certain accident sequences could result in all RHR minimum flow bypass valves being signaled to close while all other pump discharge valves also closed. The current logic configuration for Pilgrim minimum flow bypass valves is that a high flow signal from either the A or B RHR loops will close both A and B bypass valves. Thus, a postulated single failure of a flow sensing instrument may result in all RHR pumps running without flow. This condition would lead to the pumps running dead headed with potential for pump damage in a few minutes.

The design deficiency is significant to public health or safety because if this single failure occurred in conjunction with an automatic start of the RHR system, RHR pump damage may occur if unrecognized by the operator. This event could disable RHR functions including LPCI, head spray, drywell spray, shutdown cooling, torus spray and suppression pool cooling. As a result of the loss of suppression pool cooling over a long period of time, core spray pumps could ultimately lose net positive suction head and also be unavailable. Thus, systems designed to remove decay heat under both normal and accident conditions could be severely degraded.

Cause or Causes - The deficiency is attributed to error in design of the logic for RHR mini-flow control. The NRC Staff believes that the error was most likely the result of inadequate consideration of the effects of mini-flow isolation during design of logic for determining the proper (intact) coolant loop into which emergency core coolant would be injected following a loss of coolant accident.

Actions Taken to Prevent Recurrence

Licensee - Following discovery of the deficiency, all operating shifts at Pilgrim were briefed on the problem. In addition to this, the licensee performed a design review of other CSCS systems to determine if they had a similar deficiency. The results of the review confirmed that only the RHR minimum flow protection logic was deficient. Since the plant was in the cold shutdown condition at the time of the discovery of the deficiency, no other immediate mitigative actions were deemed necessary. The licensee is currently evaluating short-term and long-term modifications to correct the problem. In a May 30, 1986, letter to the NRC Staff, the licensee made a commitment to implement the short-term modifications prior to restart from the current outage.

<u>NRC</u> - The NRC Staff was first notified of the deficiency by the licensee in a report made to the NRC Headquarters Operation Center per 10 CFR §50.72 on May 19, 1986. A followup letter, pursuant to 10 CFR §21.21, was submitted on May 23, 1986. Followup discussions between the staff, licensee, and GE indicated a high likelihood that other GE BWRs in operation also had the problem.

Based on the generic applicability and the potentially significant consequences of such a failure, the NRC sent on May 23, 1986, Inspection and Enforcement Compliance Bulletin No. 86-1, "Minimum Flow Logic Problems that Could Disable RHR Pumps" (Ref. 4) to all GE BWR facilities holding an operating license or a construction permit. Recipients were required to:

1. Promptly determine whether or not their facility has this single failure vulnerability.

2. If the problem exists, immediately instruct all operating shifts of the problem and measures to recognize and mitigate the problem.

3. Within seven days of receipt of the Bulletin, provide (a) a written report to the NRC which identifies whether or not this problem exists at their facility, and (b) if the problem exists, identify the short-term modifications to plant operating procedures or hardware that have been or are being implemented to assure safe plant operations.

4. If the problem exists, provide a written report within 30 days of receipt of the Bulletin informing the NRC of the schedule for long-term resolution of problems that are identified as a result of the Bulletin.

The Bulletin noted that one of the potential fixes being proposed by GE is to remove the automatic closing signal from the RHR minimum flow bypass valves. This fix will result in some of the LPCI flow being diverted through the minimum flow line. For other RHR modes of operation, the valves may be manually closed. However, the Bulletin also cautioned that although safety analyses may justify this interim fix, there are a number of problems that need to be considered. For example, on many plants the minimum flow bypass valves must be closed during shutdown cooling in order to prevent draining the reactor vessel inventory to the torus. The minimum flow bypass valves are considered containment isolation valves on some plants.

Review of the short term responses required by the Bulletin showed that four plants other than Pilgrim also have the subject RHR minimum flow protection logic error. These plants are Quad Cities Units 1 and 2 (located in Rock Island County, Illinois) and Dresden Units 2 and 3 (located in Grundy County, Illinois). Both the Quad Cities and Dresden facilities are operated by Commonwealth Edison Company.

Short term correction actions and plans for long term correction actions proposed for the plants are currently under review by the Staff.

Future reports will be made as appropriate.

* * * * * * * *

FUEL CYCLE FACILITIES

(Other Than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the second calendar quarter of 1986. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

* * * * * * * * * * *

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 radioistopes 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 second calendar quarter of 1986. As of the date of this report, the NRC had determined that the following events were abnormal occurrences.

86-10 Willful Failure to Report a Diagnostic Medical Misadministration

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criterion) of this report notes that an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence. In addition, Example 11 of "For All Licensees" of Appendix A notes that serious deficiencies in management or procedural controls can be considered an abnormal occurrence.

Date and Place - On May 8, 1985, a patient at Mercy Hospital, Wilkes-Barre, Pennsylvania, received an injection of a radiopharmaceutical (a diagnostic dose of technetium-99m) intended for another patient. The misadministration was willfully not reported to the NRC as required by 10 CFR §35.43.

Nature and Probable Consequences - An anonymous allegation was received by NRC Region I on May 8, 1985. The alleger stated that a misadministration had occurred that morning at Mercy Hospital when the Chief Nuclear Medicine Technician mistakenly injected the wrong patient with a radiopharmaceutical. Further, the alleger stated that the misadministration would not be reported to the NRC. The required report of the misadministration was due to the NRC by July 10, 1985.

On July 17, 1985, two NRC Region I inspectors performed a routine unannounced inspection and followup of the allegation at the licensee's facility. During the inspection, the Chief Nuclear Medicine Technician stated that no misadministrations had occurred since the one reported to the NRC in 1984.

However, the inspectors noted that records showed one patient had received two radiopharmaceutical injections in a one hour period on May 8, 1985. The Chief Nuclear Medicine Technician stated that this was not because of a misadministration.

On August 7, 1985, an investigator from the NRC's Office of Investigations (OI) went to Mercy Hospital. During an interview with the Chief Nuclear Medicine Technician, she admitted she had lied to the NRC on July 17, 1985. The Chief Nuclear Medicine Technician also stated she was told that the Medical Director of Radiology, who is also the licensee's Radiation Safety Officer (RSO), did not want the misadministration reported. The RSO stated during an interview with the OI investigator on August 7, 1985, that he had informed some of his staff not to report the misadministration.

The consequences of the licensee's actions in this incident are that (1) it decreases the NRC's confidence that this licensee will report incidents as required by regulation and (2) it delays implementation of procedures to prevent further misadministrations of a similar nature.

The effects on the patient, mistakenly receiving the radiopharmaceutical, would be expected to be small due to the relatively low levels of exposure involved. However, it did represent an unnecessary exposure.

Cause or Causes - The cause is due to the deliberate failure of the RSO to follow the NRC requirements for reporting misadministrations and instructing the hospital staff not to report this particular misadministration.

Actions Taken to Prevent Recurrence

Licensee - The licensee, as well as another licensee in which the RSO is involved, requested an extension to respond to the NRC enforcement actions described below.

NRC - On June 17, 1986, the NRC forwarded to Mercy Hospital (1) an Order requiring the licensee to show cause why the Chief Nuclear Medicine Technician and the RSO should not be prohibited from the performance or supervision of any licensed activities, and (2) a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $5,000 (Ref 5).

The RSO at Mercy Hospital is also listed as an authorized user of NRC licensed material on the license of Valley Radiology Associates, Inc., Kingston, Pennsylvania. Therefore, on June 17, 1986, the NRC issued a similar Order to this licensee (Ref. 6).

Information regarding these enforcement actions was sent to all NRC medical licensees on October 3, 1986 by Inspection and Enforcement Information Notice No. 86-85 (Ref. 7).

Future reports will be made as appropriate.

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86-11 Therapeutic Medical Misadministration

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criterion) of this report notes that an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence.

Date and Place - On April 9, 1986, at Maryview Hospital, Portsmouth, Virginia, a patient received a therapy dose in a chemical form other than that intended. This resulted in an unintended dose of several hundred rads to the patient's bone marrow.

Nature and Probable Consequences - A physician asked the Nuclear Medicine Department to order a dose of phosphorus-32 as colloidal chromic phosphate on April 7, 1986, for administration to a kidney carcimoma patient for abdominal ascites reduction on April 9, 1986. This verbal order was relayed to Nuclear Medicine through a third party, and the chemical form of phosphorus-32 was not made clear. Nuclear Medicine proceeded to order 15 millicuries of phosphorus-32 as sodium phosphate because this chemical form was used more frequently than the colloidal form. The order was processed and received in the hospital in the normal manner.

On April 9, 1986, the physicist drew up the dose in a syringe, assayed it in the dose calibrator, and then put it aside. Shortly thereafter, a physician (other than the physician who ordered the dose) administered the dose intraperitoneally to the patient. Later the same day, the Chief Nuclear Medicine Technologist, while discussing this particular patient with a nurse, discovered that the soluble form, in lieu of the colloidal form of the phosphorus-32, was administered intraperitoneally. Subsequently, the radionuclide was no longer confined in the peritoneal cavity. This information was relayed to several physicians and was also reported later that day to the NRC.

On April 10, 1986, the patient was administered stable phosphorus to accelerate excretion of the phosphorus-32. Blood counts for leucocytes, red blood cells, hematocrits and platelets showed no significant depression as of April 21, 1986.

The consequences of the misadministration was a significant unintended dose to the patient's bone marrow. The licensee estimated the dose to be at least 150 rads. However, the NRC's medical consultant believes the dose could have been as much as 700-800 rads to the patient's bone marrow with an increased chance of the patient contracting leukemia.

The misadministration constituted a significant failure to comply with NRC regulatory requirements. The patient was subjected to a procedure unrelated to the authorized uses of phosphorus-32 as sodium phosphate.

Cause or Causes - The root cause was the lack of written prescriptions for ordering therapeutic doses.

Actions Taken to Prevent Recurrence

Licensee - The licensee established written procedures and forms to provide for written prescriptions and therapeutic radionuclide procedures. The licensee's agreement to establish procedures for ordering and administering therapy doses had been previously documented in an NRC Confirmation of Action Letter, dated April 10, 1986.

NRC - In addition to engaging a medical consultant and issuing the Confirmation of Action Letter, the NRC Region II conducted a special inspection at the hospital on April 11, 1986. An Enforcement Conference with the licensee was held on May 2, 1986, to discuss NRC concerns regarding the inspection findings. At the conference, the licensee presented the previously mentioned written procedures and forms.

On August 7, 1986, the NRC issued to the licensee (1) a Confirmatory Order Modifying License, and, (2) a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $2,500 (Ref. 8). The Order, effective immediately, was issued to confirm implementation of corrective procedures and to ensure their continued implementation. The licensee paid the civil penalty.

NRC Region II will review the effectiveness of the procedures during subsequent inspections.

This item is considered closed for the purposes of this report.

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86-12 Willful Failure to Report Diagnostic Medical Misadministrations

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criterion) of this report notes that an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence. In addition, Example 11 of "For All Licensees" of Appendix A notes that serious deficiency in management or procedural controls in major areas can be considered an abnormal occurrence.

Date and Place - On April 22, 1986, the NRC Office of Inspection and Enforcement issued an Order, effective immediately, removing a physician from the position of Radiation Safety Officer (RSO) and Authorized User at Bloomington Hospital, Bloomington, Indiana (Ref. 9).

Nature and Probable Consequences - On October 12, 1984, NRC Region III received an allegation that five diagnostic misadministrations had occurred at Bloomington Hospital and that they had not been reported to the NRC as required. (A diagnostic misadministration involves an error in the administration of a radioactive pharmaceutical used for a diagnostic medical test.) During a subsequent inspection, the physician serving as RSO informed the NRC that only one misadministration had occurred. After further interviews were conducted, and additional information gathered by the inspectors, the physician admitted that the other four diagnostic misadministrations did occur.

During the inspection, however, the physician obstructed the inspection and misled the inspectors by instructing hospital employees to inform the inspectors that the misadministrations had not occurred and by withholding or concealing nuclear medicine films from the inspectors.

Because of the low levels of radiation exposures to the patients in diagnostic tests, no detrimental medical effects are anticipated as a result of these misadministrations.

Cause or Causes - The NRC determined that the failure to report the misadministrations was willful.

Actions Taken to Prevent Recurrence

Licensee - As required by the NRC Order dated April 22, 1986 (Ref. 9), the licensee removed the physician from the position of RSO and as an Authorized User designated in the NRC license. Another individual on the hospital staff was placed in the position of RSO with the approval of NRC Region III.

NRC - The NRC (including the Office of Investigations) investigated the allegation and the RSO's subsequent actions described above and concluded that there was no longer reasonable assurance that the physician could be relied upon to comply with Commission requirements in the performance or supervision of licensed activities, or that the licensee would comply with Commission requirements while the physician is conducting or supervising licensed activities as an Authorized User or as the RSO at the hospital.

The license was subsequently amended to designate the new RSO.

Information regarding this enforcement action was sent to all NRC medical licensees on October 3, 1986 by Inspection and Enforcement Information Notice No. 86-85 (Ref. 7).

This item is considered closed for the purposes of this report.

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86-13 Diagnostic Medical Misadministration

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criterion) of this report notes that an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence.

Date and Place - On May 16, 1986, NRC received written notification that on May 7, 1986, an out-patient of the Robert Packer Hospital and Guthrie Clinic in Sayre, Pennsylvania, received 10 millicuries of iodine-131 rather than the prescribed radiopharmaceutical for a bone scan, technetium-99m.

Nature and Probable Consequences - Approximately two weeks before the scheduled appointment, an out-patient was mistakenly scheduled for a whole body iodine-131 scan rather than a whole body bone scan. At the time of scheduling, a verbal confirmation for an iodine-131 whole body scan was received from the doctor's office.

The patient arrived without a requisition for the study and the technician administered 10 millicuries of iodine-131 without the consultation with a radiologist required by department policy. The patient was instructed to return the following day for the imaging procedure. On return to the hospital the following morning, the patient produced an order from her physician requesting that a technetium-99m bone scan be performed. The technician proceeded to perform the whole body iodine-131 scan and then notified the radiologist of the misadministration.

The licensee informed the NRC of the misadministration and the probable medical effects were explained to the patient. The misadministration would result in a considerable dose to the thyroid. The patient was given Lugol's Solution (to help reduce the uptake of the iodine-131 by the thyroid) and instructed to take six milliliters four times per day for four days. Arrangements were made for the patient's thyroid function to be evaluated and followed.

Cause or Causes - The cause was failure on the part of a nuclear medicine technologist to adhere to department policy on the prerequisites required for radiopharmaceutical administration.

Actions Taken to Prevent Recurrence

Licensee - All concerned personnel have been retrained on the policy of not administering radioisotopes without a written requisition and of the requirement to obtain the specific consent of a radiologist for all cases requiring the administration of greater than 300 microcuries of iodine-131.

NRC - The incident was reviewed by an NRC medical consultant who concluded there was a probability of inducing hypothyroidism and that the medical care provided the individual was adequate. NRC Region I plans to review the incident as part of a routine inspection.

This item is considered closed for the purpose of this report.

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86-14 Diagnostic Medical Misadministration

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criterion) of this report notes that an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence.

Date and Place - On June 17, 1986, at the Tripler Army Medical Center, Tripler AMC, Hawaii, a patient received a dose of 3.09 mci of I-131 instead of a prescribed dose of 50 uci for a thyroid imaging procedure

Nature and Probable Consequences - A 54 year old female patient was given a 3.09 mci dose of I-131 by mistake. The patient was scheduled for a thyroid imaging procedure which utilizes only 50 uCi of I-131. The radiation exposure received by the patient due to the 3.09 mci I-131 dose is estimated to be 2472 rad to the thyroid, 0.43 rad to the ovaries, and 1.45 rad to the whole body.

Contact with the licensee was made on July 9, 1986, regarding any possible clinical symptoms or adverse health effects due to the 3.09 mci I-131 dose. The licensee stated that the patient had been hospitalized for observation. On

July 6, 1986, the patient was discharged due to the lack of any clinical symptoms. The patient has been scheduled for 90 days interval checkups at her duty station on Guam. An annual medical workup has also been scheduled. The high exposure to the thyroid may result in some degree of impairment in its function.

Cause or Causes - This misadministration was the result of an isolated incident of misreading the consultation sheet.

Actions Taken to Prevent Recurrence

Licensee - Effective immediately, the dispensing procedure for radioactive iodine is as follows:

(a) In all cases, the final dispensing and checking of the dose will be done by a staff physician or radiology resident assigned to Nuclear Medicine Service.

(b) The identification of the patient as well as the final amount dispensed will be co-signed by the physician involved in that particular procedure.

(c) The quality assurance manual for Nuclear Medicine Service is being updated to stipulate the new review procedures.

NRC - The circumstances of the misadministration were discussed in detail with the licensee on July 3, 1986 by a member of the NRC Region V management staff. The licensee's corrective actions appear to be acceptable. The NRC will not issue any further requirements in this matter at this time. The matter will be reviewed again during the next inspection.

This item is considered closed for the purposes of this report.

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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 second calendar quarter of 1986, the Agreement States reported the following abnormal occurrences to the NRC.

AS86-5 Uncontrolled Release of Krypton-85 to an Unrestricted Area

Appendix A (see the first general subcriteria) of this report notes that moderate exposure to, or release of, radioactive material can be considered an abnormal occurrence. In addition, Example 3 of "For All Licensees" of Appendix A of this report notes that 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)), can be considered an abnormal occurrence.

Date and Place - On May 8, 1985, during routine operation of a Trio-Tech "Tracer-Flo" system at Micro-Rel Division, Medtronic, Incorporated, of Tempe, Arizona, a malfunction occurred which caused approximately 11.2 curies of radioactive krypton-85 to be vented into the atmosphere.

Nature and Probable Consequences - "Tracer-Flo" systems are used to test sealed objects, such as electronic components, to determine whether they are, in fact, sealed. Tested objects are placed in a test chamber which is evacuated and a mixture of nitrogen and radioactive krypton-85 is introduced. This mixture is then removed and replaced by air. The objects are then tested for residual radioactivity. If none is detected, this would indicate that the objects have been properly sealed.

On May 8, 1985, during routine operation of the "Tracer-Flo" fine leak system, the unit "locked" into the first cycle of operation. The unit then began to run through the other cycles while maintaining the mechanical conditions of the first cycle. This situation resulted in the krypton-85 being released to an unrestricted area, rather than being retained within the system. There was no evidence that any overexposures occurred.

Cause or Causes - A thorough inspection of the machine was made and all mechanical systems were found to function properly. The failure was attributed to the machine's logic board. This was concluded by a step-by-step replacement of integrated circuits on the logic P.C. board until control panel indications were normal. The unit was then cycled a number of times and found to work properly.

Actions Taken to Prevent Recurrence

Licensee - Even though the licensee has an exemplary maintenance program, it would not have prevented this type of release. The P.C. board logic failure can only be rectified by design changes by the manufacturer.

State Agency - The Agency monitored the licensee's response to this event and confirmed completion of the actions described above. The Agency performed an inspection of the circumstances associated with the event and the licensee was assessed a civil penalty in the amount of $3,000. Due to the licensee's good past history and cooperation with the Agency, the civil penalty was mitigated to $1,500, which was imposed upon and paid by the licensee.

This item is considered closed for the purposes of this report.

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AS86-6 Contaminated Radiopharmaceutical Used in Diagnostic Administrations

Appendix A (see the general criterion) of this report notes that an event involving a moderate or more severe impact on public health or safety can be considered an abnormal occurrence.

Date and Place - On May 9, 1985, a breakthrough of molybdenum-99 (a radioactive contaminant) occurred in a molybdenum-99/technetium-99m generator at Scripps Memorial Hospital of Encinitas, California. The breakthrough went unrecognized and the contaminated technetium-99m radiopharmaceutical was administered to four patients scheduled for diagnostic medical tests. Therefore, these patients received exposures higher than necessary.

Nature and Probable Consequences - Technetium-99m is a radiopharmaceutical which is widely used in hospitals and doctors' offices for diagnosing a variety of diseases. It has a short halflife of 6 hours (i.e., it loses half of its radioactivity every 6 hours). It is a product of the decay of another radioactive material, molybdenum-99.

The technetium-99m producing devices, called generators, contain molybdenum-99. Technetium-99m, the short-lived product, is removed from the generator as needed by using a saline solution which combines with the technetium-99m, but leaves most of the molybdenum-99 in place. Molybdenum-99 has no medical application and is considered a contaminant; NRC requirements permit no more than 5 microcuries molybdenum-99 contaminant in a dose of technetium-99m.

The generator was purchased from New England Nuclear by Nuclear Pharmacy, Inc. and used to process unit doses in the San Diego, California area. Later, the generator was transferred to Scripps Hospital (a State of California licensee). (The practice of retransfers of generators for human use has since been discontinued.) After a few days of use of the generator, the licensee's nuclear medicine scanning equipment developed anomalies which made the scanning results useless (no image, but with indications of a high energy background).

After determining that the scanning equipment was not at fault, the licensee suspected molybdenum-99 breakthrough. A physician at the San Diego Veterans Administration Hospital confirmed the presence of the contaminant. He estimated liver doses to the patients ranging from 130 rads to 260 rads. As discussed further below, it is believed that DTPA was inadvertently used, rather than the saline solution, for removing technetium-99m from the generator. Therefore, due to possible rapid clearance of the DTPA from the body, the actual doses may have been less than those estimated.

Blood test results of the patients were reported to be normal, perhaps because the material may not have deposited in the vascular compartment. The whole body dose for each patient was estimated to be a few mrad. The nuclear medicine physician at the hospital reported in January 1986 that "no adverse effects have been identified in any of the four patients."

Cause or Causes - After many milkings of the generator with normal eluants, it appears that DTPA, a chelating agent, was inadvertently used in place of the usual saline solution (the vials were almost identical). This DTPA removed a substantial amount of the molybdenum-99 from the column. After the fact tests estimate that as much as 1.0 mCi/cc of molybdenum-99 may have ended up in the doses. Secondly, although molybdenum-99 breakthrough testing was routinely performed, it appears that the nuclear medicine technologist observing the dose calibrator readings had come to ignore which indicator light was lit, i.e., millicurie or microcurie and to simply record the digital readout assuming it was microcurie. There is a practical certainty that the calibrator was indicating millicuries which should have been noticed by the technologist.

Actions Taken to Prevent Recurrence

Licensee - Upon suggestion of Mo-99 breakthrough, the generator was taken out of service and affected patients were identified. The dose calibrator which had been independently checked and calibrated only one month earlier was reapproved by the licensee's consultant. All succeeding molybdenum-99 and aluminum breakthrough safety checks were confirmed by either a second nuclear medicine technician or nuclear medicine physician.

Later, the hospital discontinued the use of generators and began using bulk technetium-99m. But tests for molybdenum-99 breakthrough were continued as a precautionary measure.

State Agency - The event was investigated during an onsite visit by the Agency. The licensee was cited under one of its license conditions for failure to perform adequate molybdenum-99 breakthrough tests on the generator eluate.

This report by the Agency was considerably delayed because the Agency's medical consultant, who was asked to evaluate the patients' doses, provided vastly different (and lower) estimates than the VA hospital physician but did not provide further information to explain the discrepancies. Having received no response from the consultant to its inquiries, the Agency has accepted the VA hospital physician's dose estimates.

This item is considered closed for the purposes of this report.

REFERENCES

1. Letter from Richard W. Starostecki, Director, Division of Reactor Projects, NRC Region I, to Mr. S. L. Daltroff, Vice President, Electric Production, Philadelphia Electric Co., forwarding Inspection Report No. 50-278/86-09, Docket No. 50-278, March 25, 1986.*

2. Letter from Thomas E. Murley, Regional Administrator, NRC Region I, to Mr. S. Daltroff, Vice President, Electric Production, Philadelphia Electric Co., forwarding a Notice of Violation and Proposed Imposition of Civil Penalty, Docket No. 50-278, June 9, 1986.*

3. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 85-94, "Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA," December 13, 1985.*

4. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Compliance Bulletin No. 86-01, "Minimum Flow Logic Problems That Could Disable RHR Pumps," May 23, 1986.*

5. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to W. David Keating, Vice President, Ancillary Services, Mercy Hospital, forwarding (1) an Order to Show Cause Why the License Should Not Be Modified and (2) a Notice of Violation and Proposed Imposition of Civil Penalty, Docket No. 30-02971, June 17, 1986.*

6. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to Salvatore M. Imperiale, M.D., Director of Nuclear Medicine, Valley Radiology Associates, Inc., forwarding an Order to Show Cause Why the License Should Not Be Modified, Docket No. 30-15110, June 17, 1986.*

7. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 86-85, "Enforcement Actions Against Medical Licensees for Willful Failure to Report Misadministrations," October 3, 1986.*

8. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to J. Bland Burkhardt, Administrator, Maryview Hospital [forwarding (1) Confirmatory Order Modifying License, (2) Notice of Violation and Proposed Imposition of Civil Penalty, (3) Letter to all Medical Licensees from V. L. Miller (NRC) dated May 2, 1979, and (4) NRC Inspection Report No. 45-10831-02/86-01], Docket No. 030-03347, August 7, 1986.*

9. Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to Roland E. Kohr, President, Bloomington Hospital, forwarding an Order to Show Cause Why License Should Not Be Suspended and Modified (Effective Immediately), Docket No. 030-01644, April 22, 1986.*

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*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).

An event will be considered an abnormal occurrence if it involves 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 individual 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.

<u>For Commercial Nuclear Power Plants</u>

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).

<u>For Fuel Cycle Licensees</u>

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 April through June 1986 period, the NRC, NRC licensees, Agreement States, Agreement State 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. The updating provided generally covers events which took place during the report period, thus some information is not current. Some updating, however, is more current as indicated by the associated event dates. Open items 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. 9, No. 1. It is further updated for this report period.

Reactor Building Entries

During the second calendar quarter of 1986, 91 entries were made into the TMI-2 reactor building, bringing the total number of entries since the March 1979 accident to 956. Reactor building activities during this period centered around (1) controlling microorganism populations in the reactor coolant system (RCS) and thereby improving water clarity, (2) pick and place defueling, and (3) the use of a drilling rig to obtain stratified samples of the damaged core. Additional reactor building entries were made to remove two of the six internal vent valves and perform maintenance on systems needed for the cleanup effort.

Reactor Vessel Defueling Operations

As discussed in the previous update report, visibility in the reactor vessel progressively decreased during the first quarter of calendar year 1986 to a point at which defueling activities were being adversely affected. Microorganisms (algae, fungi, bacteria, and anaerobes) were initially identified as the cause of filter plugging in the Defueling Water Cleanup System which rapidly made this system become ineffective in maintaining water clarity. The licensee conducted a three phase program of chemical treatment, high pressure flushing and filtration of the RCS. At the end of April 1986 the Temporary Reactor Vessel Filtration System, using a large diatomaceous earth filter, was put into operation. High pressure hydrolancing of the RCS began in early May 1986. Also in May, the licensee began injecting borated water treated with hydrogen peroxide into the RCS via the Standby Pressure Control System. The licensee established an initial 200 ppm residual concentration of peroxide as a biocide to kill the microorganisms. As a result, water visibility was improved to over one foot. The licensee is continuing the program on a periodic basis to control the populations of microorganisms and maintain water clarity.

Pick and place defueling (using remote tools to move damaged fuel and core debris into defueling canisters), halted during the microorganism control program, resumed on May 23, 1986. Near the end of June, pick and place defueling was again halted to ready the work platform for installation of the drilling rig to obtain stratified core samples. One additional canister was transferred to the fuel pool at this time bringing the total number of canisters transferred out of the reactor building to 43. The total weight of debris removed from the reactor vessel thus far is 51,670 lbs., representing nearly 17% of the 308,000 lbs. of debris estimated to be in the vessel.

Core boring to obtain stratified samples of the core began on July 3, 1986. This effort was designed to obtain full length samples of the reactor core from the surface of the debris bed to within inches of the inner surface of the lower head of the reactor vessel. Information gained from drilling regarding the hardness, ductility, and friability of the core material will be used in planning future defueling activities. The core samples (approximately 2 1/2" in diameter and 8 ft. long) will be shipped in canisters to the Idaho National Engineering Laboratory (INEL) for examination.

Dose rates associated with defueling activities have continued to remain low. Dose rates on the defueling work platform average approximately 8 mrem/hr, and the highest measured dose rates during canister transfer from the reactor vessel to the storage racks have been less than 40 mrem/hr.

EPICOR-II Submerged Demineralizer System (SDS) Processing

Approximately 212,474 gallons of water were processed through the SDS during the reporting period. Approximately 148,468 gallons were processed through the EPICOR-II system during the quarter.

Cask and Liner Shipments

There were no offsite shipments of EPICOR-II or SDS liners. At the end of June, seven defueling canisters stored in the fuel storage pool were loaded into the first rail-mounted shipping cask. The total weight of the debris contained in the seven canisters is 2400 lbs. The first shipment is expected to occur sometime in July 1986. Additional shipments are expected over the next two and one-half years.

Auxiliary and Fuel Handling Building

Decontamination activities continued during the second quarter of 1986. Activities included: scabbling and painting of the floors in the neutralizer tank rooms and the mini decay heat pump room; steam cleaning of the tendon access gallery; scabbling in the reclaimed boric acid tank room; vacuum cleaning of the 281' elevation; and water flushing of the Westinghouse valve room.

TMI-2 Advisory Panel Meetings

The Advisory Panel for the Decontamination of Three Mile Island Unit 2 met on April 10 1986, in Harrisburg, Pennsylvania and June 11, 1986, in Washington, D.C.

At the April 10, 1986, meeting, the Panel was briefed by a representative of the licensee on the status of defueling. The panel expressed interest on the issue of microorganism growth in the reactor coolant system. The licensee described the extent of the problem and presented plans for control of the populations. The NRC Staff provided a status report on regulatory issues related to TMI-2. This included a brief summary of the NRC Advisory Committee for Reactor Safeguards (ACRS) conclusions on the potential for recriticality of the TMI-2 core during defueling.

On June 11, 1986, the Panel met with the NRC Commissioners in Washington, D.C. The Panel expressed satisfaction with the ACRS review of the recriticality issue. The Panel informed the Commission of the continuing interest and concern of the local citizens regarding the TMI-2 accident related health issue. The Panel also reported to the Commission general approval of the Department of Energy's plans for the offsite shipment of the damaged TMI-2 fuel. The Commission and the Panel discussed proposed Panel activities for the near future and also addressed the question of the point at which Panel activities will conclude.

Future reports will be made as appropriate.

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80-5 Decay Heat Removal Problems

This item, originally titled "Loss of Decay Heat Removal Capability," involving Davis-Besse Unit 1, was reported and closed out in NUREG-0090, Vol. 3, No. 2, "Report to Congress on Abnormal Occurrences: April-June 1980." It is being reopened, and a more generalized title used, in order to describe numerous other similar events which have occurred at many of the U.S. pressurized water reactors (PWRs).

As described in the original report, the Davis-Besse plant experienced nine losses of decay heat removal (DHR) capability during 1980. The basis for reporting the events as an abnormal occurrence was a serious deficiency in management or procedural controls in major areas.

Subsequent to the Davis-Besse events, the NRC's Office for Analysis and Evaluation of Operational Data (AEOD) and others began an analysis of DHR system losses at all U.S. PWRs. The results of the AEOD analysis were issued during December 1985 as Case Study Report AEOD/C503 (Ref. B-1). The report covered the period from 1976 through 1983, and part of 1984.

As described in the Case Study Report, the DHR system [also referred to at various plants as the residual heat removal (RHR) system, and shutdown cooling (SDC) system] is designed to remove fission product decay heat from the reactor core. The safety function of the DHR system is to remove heat from the primary system at a rate that will enable operators to bring the plant from hot shutdown conditions to cold shutdown or refueling conditions, and to maintain the plant in such shutdown conditions for extended periods of time. For the transition phase associated with cooling the plant from operating pressures and temperatures after a reactor trip, for example, to hot shutdown, the steam generators and the auxiliary feedwater system are used to remove heat from the primary system. Upon reaching the reduced pressures and temperatures associated with the hot shutdown condition, the DHR system is activated. Most DHR systems operate at temperatures of 350°F or less, and at pressures less than 600 psig.

Though there are design differences among the various plants, a typical DHR system is composed of two redundant 100% capacity trains, usually located outside containment. Water is taken from a reactor coolant system (RCS) hot leg, flows progressively through suction isolation valves, the DHR pump, the DHR heat exchanger and back to the reactor vessel. During accident conditions, most DHR systems can be aligned to provide low pressure emergency core coolant functions.

Depending upon plant and reactor conditions, an extended loss of the DHR function could lead to bulk boiling, core uncovery, and possible fuel damage unless timely operator actions are taken either to restore the DHR function or to provide some alternate means of cooling water to the core. The time available for restoring cooling water prior to core uncovery can be short (i.e., an hour or less) or long (i.e., up to many hours, weeks, or years) depending on the conditions at the time the DHR system is interrupted.

The time margin available for restoring the DHR system, or establishing alternate methods of heat removal, depends upon the RCS temperature, the decay heat rate (which is dependent upon time interval elapsed from reactor trip to DHR system failure and core power operating history), the amount of RCS inventory, and the availability of backup sources of borated water. During some shutdown operations, the RCS may be partially drained (e.g., to perform steam generator inspections or repairs). Decreased primary system inventory can significantly reduce the time available to recover the DHR function prior to bulk boiling and core uncovery.

The AEOD study found that from 1976 through 1983, at least 130 loss-of-DHR system events had been reported to the NRC by PWR licensees. There also were others which were not reportable under the NRC reporting requirements in effect prior to January 1, 1984. At the time of preparing the AEOD report, the 1984 data were incomplete; however, there were at least 10 events in 1984 which were considered significant.

For all cases, plant personnel were able to restore the DHR function prior to reaching an unsafe condition (i.e., core uncovery). However, with an extended delay in restoring the DHR function, some of the loss of DHR events could have led to fuel failure and more serious radiological consequences.

While none of the recorded DHR failures affected the health and safety of the public, some of the events caused significant plant disruptions, extended downtime, and expensive cleanup and recovery.

The AEOD study concluded that the underlying or root causes of most of the loss-of-DHR system events were human factors deficiencies involving procedural inadequacies and personnel error. Most of the errors were committed during maintenance, testing, and repair operations.

The categorization of the 130 DHR system failures that occurred at PWRs during the years 1976-1983 is shown in Table B.1.

Table B.1

Categories of 130 Reported Total DHR System Failures When Required to Operate (Loss of Function) at U.S. PWRs 1976-1983

	No. of Events	(% of Events)
Automatic Closure of Suction/ Isolation Valves	37	(28)
Loss of Inventory	36	(28)
· Inadequate RCS Inventory Resulting In Loss of DHR Pump Suction (26)		
· Loss of RCS Inventory Through DHR System Necessitating Shutdown of DHR System (10)		
Component Failures	29	(22)
· Shutdown or Failure of DHR Pump (21)		
· Inability to Open Suction/Isolation Valve (8)		
Others	28	(22)
Total	130	(100.0)

Events involving problems with the suction/isolation valves and the DHR pumps accounted for about 71% of the DHR system failures. Automatic closure of the suction/isolation valves accounted for about 28% of the events. (These valves are located in the flow path from the reactor coolant system hot leg(s) to the suction side of the DHR pumps.) The underlying or root causes of most of these events were human errors.

About 28% of all reported DHR system losses involved loss of RCS inventory or inadvertent reduction of reactor vessel water level below the hot leg outlets. Twenty-six of these events resulted in inadequate pump suction, cavitation, or air binding. Many events of this type were significant because of their long recovery times. Recovery required refilling the RCS and bleeding off the air of vapor bound pump(s).

About 22% of the reported DHR losses involved DHR system component failure. Twenty-one events involved shutdown or random failure of an operating DHR pump when the other pump or train was inoperable. Eight events involved previously closed suction/isolation valves that could not be opened.

Corrective actions taken by licensees have generally consisted of various combinations of equipment modifications, changes in procedures and maintenance, and training of personnel. For some plants, the corrective actions taken have not significantly reduced the occurrence of DHR losses, or the duration of the losses. Other plants, such as Crystal River and Davis-Besse, have shown considerable improvement starting about the same time that these licensees implemented actions to improve their planning, coordination, and management of outage and maintenance activities.

The problems of DHR failures have been under review for some time by the NRC Office of Nuclear Reactor Regulation (NRR). Under Unresolved Safety Issue (USI) A-45, the review is directed toward determining the need for improvements in current DHR systems. Under Generic Issue No. 99, the review is specifically directed to loss of the RHR system during cold shutdown or refueling. Both USI A-45 and Generic Issue No. 99 have been assigned high priority for resolution.

The AEOD Case Study Report included several recommendations based upon the potential safety significance of loss-of-DHR events for significantly improving DHR system reliability and availability. The recommendations included improving human factors by upgrading coordination, planning, and administrative control of surveillance, maintenance, and testing operations which are performed during shutdown; providing operator aids to assist in determining time available for DHR recovery and to assist operators in trending parameters during loss-of-DHR events; upgrading the training and qualification requirements for operations and maintenance staff; requiring the use of reliable, well-analyzed methods for measuring reactor vessel level during shutdown modes; modifying plant design to remove autoclosure interlocks and/or power to the DHR suction/isolation valves during periods which do not require valve motion; and clarifying plant technical specifications to eliminate ambiguities associated with operating mode definitions.

NRR has included the AEOD recommendations in their review under Generic Issue No. 99.

Further review of Licensee Event Reports submitted to the NRC since the end of 1983 shows that loss of DHR events continue to be a problem area. There were about 15 events during 1984, 18 during 1985, and 7 as of the end of May 1986.

One of the more significant events during this period occurred on March 26, 1986, at San Onofre Unit 2 during a refueling outage. Even though the plant had been shut down for 11 days, there still was substantial fission product decay heat. The reactor water heated up faster than plant personnel expected (the operators believed boiling would not begin before one hour). During the event, the DHR system was unavailable for only 49 minutes. Nevertheless, the reactor water was heated to boiling within about 40 minutes after the total loss of DHR flow. (This time agrees well with that shown in Figure 4, "DHR Recovery Time Margin" in the AEOD Case Study Report.) Boiling continued for about 7 minutes until DHR flow was reestablished. Core uncovery could have begun within about another hour if no water addition was made. Radioactivity levels in the fuel handling building increased slightly. The levels did not reach hazardous levels, however, the event did represent an unanticipated risk to plant personnel. In addition, the event might not have occurred had a level indication problem, which occurred earlier the same day, been properly analyzed and corrected. Details on the event are discussed below. Description of the event also serves to illustrate one of the major causes of DHR system loss previously discussed (i.e., inadequate RCS inventory resulting in loss of DHR pump suction).

The plant was shut down on March 15, 1986, for a refueling outage. At 9:50 p.m. (PST) on March 26, 1986, the DHR pump became inoperable due to entrapment of air. The entrapped air had been pulled into the pump from the DHR suction line which connects to the bottom of the RCS hot leg discharge piping a few feet from the reactor vessel. At the time of the event, the RCS was depressurized and vented to atmosphere and the reactor vessel water level had been purposely drained to a level of the middle of the hot leg. In this condition the horizontal reactor coolant system piping hot leg is half full of water and half full of air.

This condition provides adequate flow path and suction head for the DHR pumps, with water levels sufficiently low to permit removal of steam generator primary side access covers for tube inspections. Slightly lower levels are known, from previous experience, to be susceptible to creating an air/water vortex which can pull air into the DHR suction line at the bottom of the reactor coolant hot leg, possibly binding the pump with air. Because of this susceptibility, the reactor water level and the DHR flow rate need to be carefully controlled when in this condition.

At the time, the operators were unaware that the reactor vessel water level indication was faulty; the level was, in fact, 10 inches lower than they believed. Therefore, when the operators lowered the water level 12 more inches to support a maintenance request, the motor amperage on the DHR pump began to take large swings indicating air entrapment. The reactor coolant temperature was 115°F at this time. The operators waited three minutes and restarted the pump. In a few minutes, the same amperage swings recurred and the pump was stopped. The standby pump was started with similar results.

The operators then followed their operating procedures for abnormal plant conditions and sent equipment operators to vent the DHR piping outside the containment penetration area.

Forty-nine minutes after the total loss of DHR flow, the venting was completed and the main DHR pump was restarted. The DHR hot leg temperature increased momentarily, during this period, to 210°F, and then dropped to below 200°F. However, the temperature indicators are located above the middle of the hot leg; therefore, they were not immersed in water and may not have responded in a timely way.

At this time, the ventilation system in the fuel handling building automatically isolated due to high count rate on radiation monitors. This was due to noble gas being released into the containment via the reactor vessel head instrument nozzles, and being drawn through the fuel transfer tube by the lower pressure in the fuel handling building. The noble gas came from previously damaged fuel elements (there was no evidence of any fuel damage due to the March 26, 1986 event) and was released from the RCS due to the vessel water temperature increase, and boiling, during the event.

There were no significant effects on the health or safety of the public due to the event. RCS water addition could have been made at any time during the loss of SDC event. A high pressure safety injection pump was used to maintain RCS water level after RCS flow was reestablished. Additional flow paths of RCS makeup water were also available including gravity drain from the refueling water storage tank and safety injection tanks (which were still filled). There was no fuel damage due to the event. Radioactivity released was only about 2 curies (primarily xenon-135), which would result in a total exposure of an individual at the Exclusion Area Boundary of less than 0.002 mrem.

The primary causes of this event were inadequate level indication equipment installation, calibration, and operation; inadequate shift turnover; insufficient operator sensitivity to the importance of reactor water level control; inadequate procedures; and inadequate training. Additionally, a contributing factor was the relative ease of inducing vortexing and subsequent false level indications inherent in the unit's design.

The licensee had installed a new reactor vessel water level indication system for this refueling. The system consisted of two electrical level indicators, a wide range and a narrow range indicator. Apparent problems with the system were noted on March 19, 1986, during initial draining of the pressurizer. The problems required recalibration, apparently due to design data errors. Other level accuracy problems were observed during the following few days and on March 22, 1986, the licensee installed the level indication system used during the previous refueling, a tygon tube standpipe.

Subsequent to the event, it was determined that the tygon tube had been mounted on an old structural stanchion which had reactor water levels marked with a felt tip pen rather than on an engineered scale installed for that purpose. Secondly, the tygon tube was installed with an air bubble. The scale used was in error by two inches. The air bubble's effect on accuracy varied dependent upon the bubble's location. The operators were not aware that the tygon tube would indicate water levels that were significantly in error; therefore, they relied on it to verify proper water level in the reactor vessel.

Despite these problems, the licensee had an opportunity to discover the improper water level indication and take corrective actions to preclude the total loss of RHR system event. Earlier in the day of the event, the day shift had begun drain-down, noted that the tygon and electrical level indications were not in agreement, and had noticed the RHR pump amperage start to oscillate (indicating vortexing). The day-shift immediately added water until indications of air entrapment ceased. These occurrences were not communicated to the on-coming shift, the shift that later experienced the loss of shutdown cooling, because the out-going shift did not believe them to be significant in relation to plant conditions at the time of shift turnover.

The licensee has taken, or will take actions to establish a more reliable indication of water level. This is to be done by classifying the systems as safety-related and applying the more stringent associated administrative controls; by controlling the configuration of the level indicating systems with more detailed design documentation; by improving procedures for installation; and by establishing criteria for verification of operability. Additionally, the licensee is reconsidering a design change to make the RHR pump self-venting. Consideration is being given to developing a correlation of shutdown cooling flow and reactor water level to identify vortexing regions. Further, consideration is being given to extensive operator training concerning RHR operation and to operational procedure improvements. Implementation of several of these improvements and a significant increase in management and operator control were observed during a subsequent drain-down to reestablish mid-loop conditions on April 22, 1986.

Upon being notified of the event, the NRC Resident Inspectors began an examination of the event, including the root causes. The inspection report, together with a Notice of Violation, were sent to the licensee on June 20, 1986 (Ref. B-2). NRC Region V will continue to review the licensee's corrective actions.

85-1 Premature Criticality During Startup

This abnormal occurrence, which occurred at Summer Unit 1 on February 28, 1985, was originally reported and closed out in NUREG-0090, Vol. 8, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1985."

The item is being reopened to describe two somewhat similar events which occurred on July 1, 1985, at Fermi Unit 2 and on April 13, 1986, at San Onofre Unit 3. Fermi Unit 2 is a General Electric-designed boiling water reactor, operated by Detroit Edison Company, and located in Monroe County, Michigan. San Onofre Unit 3 is a Combustion Engineering-designed pressurized water reactor, operated by Southern California Edison Company, and located in San Diego County, California.

The Fermi Unit 2 event was briefly mentioned as part of Abnormal Occurrence 85-20 ("Management Deficiencies at Fermi Nuclear Power Station") in NUREG-0090, Vol. 8, No. 4 ("Report to Congress on Abnormal Occurrences: October-December 1985"). A more detailed description of the event, together with the significant enforcement action proposed, are discussed below.

Fermi Unit 2 Event

On July 1, 1985, a reactor operator (the Nuclear Supervising Operator at the control panel), about an hour into his shift, while withdrawing control rods to achieve criticality on his first attempt ever to bring a commercial power reactor critical, pulled 11 rods in Group 3 to the fully withdrawn position (position 48), rather than position 04 required by the rod pull sheet. This resulted in the reactor prematurely reaching criticality although this was not fully recognized by the licensee until several days later.

While pulling the 11th control rod in Group 3, the Short Period Alarm annunciated five times and the pen for the Channel A Source Range Monitor failed to ink for about three minutes. When the pen began inking again the count rate was increasing. At about the same time, the rod pull error was recognized and the reactor operator began reinserting the 11 rods. The Nuclear Shift Supervisor (NSS), was called and came out of his office to consult with the reactor operator. The NSS, who was also responsible for directing his first startup of a commercial power reactor, reviewed the event with the reactor operator and Shift Technical Advisor in Training and determined that the reactor had not gone critical. He then authorized recommencing of rod pulls. In making this decision he did not consult with the Shift Reactor Engineer, the Shift Operations Advisor or the Shift Technical Advisor. However, after the recommencement of rod pulling he did contact the Operations Supervisor at his home and briefly discussed the event with him.

During the event, both the NSS and the Nuclear Assistant Shift Supervisor were in the NSS's office. While the rod pull error was documented in a number of records, it was not recorded in the control room log or the Shift Supervisor's log. The Nuclear Supervising Operator in charge of the control room, with responsibility for the control room log, did not learn of the event until after the shift was over. Neither the Shift Operations Advisor nor the Shift Technical Advisor was observing the rod pulling nor was aware of the incident at the time it happened.

A Shift Reactor Engineer made the determination on July 4, 1985, that the reactor had been critical on July 1, 1985, with a 114 second period, and informed his management. Several licensee meetings were held on July 5 and 6, 1985, to discuss the event and to initiate an investigation into its cause. The NRC Senior Resident Inspector was informed of the premature criticality on July 15, 1985, and notified NRC Region III.

On July 16, 1985, NRC Region III issued a Confirmatory Action Letter (Ref. B-3) to the licensee, confirming the licensee's agreement not to operate the unit above five percent power until the premature criticality incident was fully analyzed and corrective action taken. Operation above five percent power would not occur without authorization from NRC Region III. That restriction was not lifted prior to the outage which began on October 10, 1985.

The NRC's inspection and review of this event identified nine apparent violations of NRC requirements. The results of the inspection were forwarded to the licensee on January 7, 1986 (Ref. B-4).

The event did not result in any actual safety consequences. However, it was of serious concern because it represented a serious breakdown in the management controls and discipline in the control room. Therefore, on July 3, 1986, the NRC forwarded to the licensee an Order Modifying the Licensee Effective Immediately, and a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $300,000 (Ref. B-5).

The Notice of Violation and Proposed Imposition of Civil Penalties pertained to (1) failure to follow the rod movement procedure and to properly document the position of control rods during startup; (2) licensee management did not adequately supervise the startup activities, shift turnover procedures were inadequate, and the control rod problem was not properly documented; and (3) the plant resumed reactor startup activities without fully evaluating the problem of the control rod movement which had led to the premature criticality.

The Order required the licensee to institute a program to audit control room activities to ensure that they are conscientiously carried out. The Order also requires retraining of the shift supervisor who was on duty at the time of the premature criticality, if the supervisor is to be returned to control room duties.

This item is considered closed for the purposes of this report.

San Onofre Unit 3 Event

At 11:16 a.m. on April 13, 1986, while the licensee was performing a recovery from a spurious reactor trip which had occurred the previous day, the reactor tripped after achieving criticality at a withdrawal position lower than predicted by the Estimated Critical Position (ECP). The licensee made a 4-hour report of the reactor trip to the NRC Duty Officer at 3:00 p.m. PST. This Emergency Notification System (ENS) report indicated that the reactor had tripped on departure from nucleated boiling ratio (DNBR) and local power density (LPD) trip signals generated by the core protection calculator (CPC), because of penalty factors resulting from minor control rod misalignments. The following sequence of events occurred:

- Operations personnel calculated an ECP of 60 inches on group 6, based on an estimated criticality at 11:00 a.m. Although two previous startups had been conducted since the first refueling outage which ended in early 1986, this was the first startup with significant xenon present.

- Regulating groups 1, 2, and 3 were withdrawn to their upper group limits. However, they were not individually aligned to uniform height at the fully withdrawn position, as specified in the procedure, because of operators' preoccupation with achieving criticality near the 11:00 a.m. time on which the ECP was based. As a result, some group 1 rods were inserted approximately one inch more than group 2 rods, which generated an out-of-sequence penalty factor signal to the CPC. The operators' intent was to dress the rods after criticality was achieved before the penalty factors became effective (at 10E-4% power).

- When group 4 rods reached approximately 80 inches withdrawn, the reactor achieved criticality (at approximately 10E-5% power). This was not immediately recognized by the operator and rod withdrawal continued to approximately 114 inches before the critical condition was recognized. The reactor controls were being manipulated by a reactor operator trainee under the direct supervision of a licensed reactor operator. Other licensed personnel were also present in the control room.

- When criticality was recognized, the operator began inserting control rods. However, power had by this time reached 10E-4% power, at which point the CPC reactor trips were enabled. Increasing penalty factors as the rods were inserted initiated DNBR and LPD reactor trips at a rod position of 98 inches on group 4. Actual unsafe DNBR or LPD conditions did not exist. Peak reactor power during the transient was approximately 10E-2%, with startup rates in the range of 1.5 to 1.7 decades per minute.

The licensee determined on April 13, following the trip, that the incorrect ECP was caused by incorrect xenon tables (used to predict the reactivity worth of xenon) in the Operations Physics Data Summary Book. Due to improper administrative control of the ECP data, the xenon tables for cycle 1 operation were being used instead of the tables required for cycle 2. Indications are that the licensed operator who was performing the startup (i.e., supervising the trainee) also did not devote appropriate attention to available indications of approaching criticality and was not anticipating criticality "at any time" as directed by the approved procedure.

Following a post-trip review by the licensee and an appropriate addition of boron, Unit 3 started up later on April 13 and was operating at full power on April 14.

On June 20, 1986, a Notice of Violation was transmitted to the licensee for failure to closely monitor nuclear instrumentation and for failure to recognize the premature criticality.

NRC Region V reviewed the circumstances associated with the event during their routine inspection of San Onofre Units 2 and 3 operations from March 28 through May 12, 1986. The inspection report, together with a Notice of Violation, was forwarded to the licensee on June 20, 1986 (Ref. B-6). The violation, associated with this event, pertained to the licensee's failure to closely monitor nuclear instrumentation while starting up the reactor and for failure to recognize the premature criticality.

This item is considered closed for the purposes of this report.

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85-7 Loss of Main and Auxiliary Feedwater Systems

This abnormal occurrence, which occurred at Davis-Besse on June 9, 1985, was originally reported in NUREG-0090, Vol. 8, No. 2, "Report to Congress on Abnormal Occurrences: April-June 1985," and updated in NUREG-0090, Vol. 8, No. 3; Vol. 8, No. 4, and Vol. 9, No. 1. It is further updated as follows.

As mentioned in the previous update, based upon the findings of the NRC Incident Investigation Team reported in NUREG-1154 (Ref. B-7) the NRC identified the concerns Toledo Edison Company (the licensee) should address for NRC review before resumption of operation of the plant can be approved. These concerns were identified to the licensee in a letter dated August 14, 1985 (Ref. B-8).

The licensee has responded in a document submitted to the NRC on September 10, 1985, entitled, "Davis-Besse Course of Action" (Ref. B-9). The NRC Staff has completed its review of this document and has issued a Safety Evaluation Report (NUREG-1177) addressing plant re-start (Ref. B-10). The Staff's findings were presented to an ACRS subcommittee on June 27, 1986.

The licensee is conducting its System Review and Test Program in which 34 safety systems are being extensively reviewed, including an evaluation of the system design requirements. Previous surveillance tests of the systems are being analyzed and additional testing is being performed to demonstrate the operability of the systems. The testing program includes 172 existing or modified surveillance tests and 106 new test procedures.

On March 16, 1986, the licensee identified possible cracks in the shafts of the reactor coolant pumps. There are four pumps - two associated with each of the two steam generators. One shaft was replaced with a spare shaft and the original shaft was sent to the Babcock and Wilcox Research Center for examination. Physical examination of the shaft did not reveal any cracks in the shaft. However, some cracks were observed in the bolts which connect the shaft to the pump impeller. In light of this, the licensee decided to replace the remaining shafts with new assemblies. All four shafts have been installed and the pumps are undergoing testing.

Other work required prior to the unit restart is continuing. This effort includes the replacement or repair of improperly installed Raychem electrical terminations, a problem which was identified by the licensee.

Editor's Note

During January 1986, the NRC Commissioners established an independent Ad Hoc Group to review issues subsequent to the complete loss of feedwater event at Davis-Besse on June 9, 1985, including the NRC Incident Investigation Team (IIT) investigation of that event. The Commission asked the Group to identify additional lessons that might be learned and from these to make recommendations to improve NRC oversight of reactor licensees. To fulfill its charter, the Ad Hoc Group examined the following: (1) pre-event interactions between the licensee and NRC concerning reliability of the auxiliary feedwater system and associated systems; (2) pre-event probabilistic assessments of the reliability of plant safety systems, NRC's review of them, and their use in regulatory decision-making; (3) licensee management, operation and maintenance programs as they may have contributed to equipment failures and NRC oversight of such programs; and (4) the mandate, capabilities of members, operation, and results of the NRC Davis-Besse IIT, and the use to which its report was put by the regulatory staff.

Based on the Ad Hoc Group's investigations, numerous conclusions and recommendations for improving NRC's oversight of all reactor licensees were developed. These conclusions and recommendations are discussed in detail in NUREG-1201 which was published during June 1986 (Ref. B-11). NRC management is reviewing and implementing the recommendations as appropriate.

Future reports will be made as appropriate.

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85-14 Management Deficiencies at Tennessee Valley Authority

This abnormal occurrence was originally reported in NUREG-0090, Vol. 8, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1985," and updated in Vol. 9, No. 1. It is further updated for the report period.

The NRC Staff, led by a senior management team, has identified a number of major Tennessee Valley Authority (TVA) issues requiring resolution prior to the restart of any of the TVA reactors and has provided periodic status reports to the Commission, the most recent of which was issued September 12, 1986. The staff last briefed the Commission on these issues on June 6, 1986, and stated that Sequoyah Unit 2 was expected to be the first reactor ready to resume operation. TVA last briefed the Commission on their activities on March 11, 1986.

TVA has publicly stated that restart of Sequoyah Unit 2 is estimated to be in March 1987. Unit 1 is expected to follow shortly thereafter. TVA's revision to the Sequoyah Nuclear Performance Plan was submitted in late July and is currently under review. Watts Bar is not expected to be ready for licensing before May 1987. Restart dates for the Browns Ferry Units remain uncertain.

There are seven areas where there has been considerable TVA activity and which have also received a significant level of Staff attention. These areas are corporate activities, equipment qualification, employee concerns, welding, design control, quality assurance, and licensed operator requalification. The following paragraphs summarize activity during the second quarter CY 1986 in these areas.

Corporate Activities

In January 1986, after the arrival of the new Manager of Nuclear Power (Mr. Steven White), TVA announced a major management reorganization and restructuring which was intended to address the existing deficiencies in management and employee relations. The details of this plan were presented in the revised Corporate Nuclear Performance Plan. TVA provided a second revision to the plan on July 17, 1986.

The ACRS Ad Hoc Subcommittee on TVA met with TVA and the Staff on June 12-13, 1986, to discuss the Corporate Nuclear Performance Plan and receive a briefing on the root causes of the current situation and corrective actions underway. The full ACRS met with TVA and the Staff on July 10, 1986. ACRS concerns include training of TVA managers, the transition from contractor managers back to TVA employee managers, and the span of control of the Manager of Nuclear Power.

The Commission's evaluation of intimidation and harassment (I&H) at TVA was forwarded to TVA on June 2, 1986. The Staff is reviewing all TVA I&H issues that were received prior to March 1986 to determine if they are being tracked and followed up as appropriate by the TVA General Counsel, Inspector General, or Commission staff. The TVA Inspector General (TVA IG) has expressed a need to contact concerned individuals directly, as part of his investigative process. The Staff (including the NRC Office of Investigations) is developing a method to contact selected concerned individuals in an attempt to establish communications between these individuals and the TVA IG.

Equipment Qualification

The TVA certification of completion of environmental qualification actions at Sequoyah has not been received. As TVA has completed portions of the equipment qualification (EQ) effort, the Staff continues to conduct onsite reviews.

Discrepancies identified by both the Staff and TVA at Sequoyah remain to be corrected; however, the TVA EQ Program appears to be sound. The Staff believes that the major EQ discrepancies which need to be resolved by TVA before they will be able to certify that Sequoyah is in compliance with the EQ rule involve Raychem electrical cable splices and the environmental effects of certain high energy line breaks.

TVA and the Staff will face additional effort to assure EQ compliance at Browns Ferry and Watts Bar.

Employee Concerns

Nearly 5,000 TVA employee concerns have been raised at the Watts Bar facility. Some of these involve safety-related equipment or intimidation, harassment, or wrongdoing issues. About 400 of these concerns apply to the Sequoyah facility.

TVA has established an internal program for evaluating employee concerns to replace the former program managed by Quality Technology Company (QTC). On August 29, 1986, TVA provided the NRC with a detailed description of their new program for dealing with employee concerns. The Staff will evaluate (on a sampling basis) the TVA resolution to these concerns during the coming months.

TVA and QTC terminated their contract in April 1986 at which time the Staff obtained copies of all of the QTC-generated employee concerns records. During the second quarter of CY 1986, the Staff has nearly completed the screening and expurgation effort on the QTC employee concerns files to identify safety-related information considered by NRC as needed by TVA to resolve individual concerns. This information is being transmitted to the TVA IG for further screening prior to release to the TVA line organizations. To date, the Staff screening has identified about 100 potentially new safety-related issues.

TVA has regrouped the Watts Bar employee concerns to facilitate evaluation and resolution. In early June, the Staff inspected TVA progress. Based on limited review, the Staff remains concerned about the details and depth of the TVA evaluation process. The Staff has developed review guidelines and made personnel assignments to support NRC technical review of the resolution process.

Welding

Staff review of welding at Sequoyah is nearing completion and TVA's actions appear to be acceptable. However, TVA needs to respond to the Staff's questions and approve the related employee concerns investigation reports in order for the Staff to complete its review.

TVA continues to reinspect welds at Watts Bar. In late June, the NRC Staff met with TVA to discuss their program. While the Staff considers the logic of the overall program to be basically sound, the docketed program is deficient in that it contains insufficient detailed information to permit an adequate technical review. Some of the key information needed includes sample size and acceptance criteria, applicability of portions of the ASME Code, certain quality assurance aspects related to welding, and detailed project procedures. NRC inspection of this activity is ongoing and will focus on:

- Independent examinations of hardware
- EG&G inspector qualification/certification records
- TVA's engineering calculations
- Implementation of EG&G's reinspection efforts

Design Control

TVA is developing their program for evaluating the adequacy of design control at Sequoyah. The design control program was submitted as part of the July 17, 1986 revision to the Sequoyah Nuclear Performance Plan. Inspection of TVA progress in this area was conducted in June 1986 and periodic inspections are planned in the coming months. This may be a pacing item for Sequoyah restart.

Quality Assurance

Quality assurance (QA) issues involve, primarily, TVA construction activities at Watts Bar. Numerous allegations regarding the QA program have been raised and are under review by TVA and the Staff. The Staff is reviewing a May 1, 1986 revision to the TVA QA Topical Report and met with TVA on this report in early July.

Licensed Operator Requalification

In response to the unsatisfactory performance of Browns Ferry operators on the NRC-administered requalification examination given in November 1985, TVA has implemented an extensive program to upgrade Browns Ferry licensed operator knowledge and capabilities. The first group (of three groups) completed training in June. The new retraining program appears to be acceptable to the Staff and implementation of the program is being evaluated.

The Staff is currently conducting normal inspection activities at all of the TVA facilities and conducting special inspections or reviews of particular TVA issues, as described above.

Future reports will be made as appropriate.

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85-20 Management Deficiencies at Fermi Nuclear Power Station

This abnormal occurrence was originally reported in NUREG-0090, Vol. 8, No. 4, "Report to Congress on Abnormal Occurrences: October-December 1985," and updated in NUREG-0090, Vol. 9, No. 1. It is further updated for the report period.

During the maintenance and modification outage, which began October 10, 1985, the licensee identified several engineering and design issues, including the handling of design modifications performed between 1984 and 1986, the status of seismic design calculations, and the potential overloading of concrete embedments. These issues were resolved by the licensee and its consultants during the outage. The handling of the information relating to these engineering issues, however, remains under investigation by the NRC Office of Investigations.

Other significant equipment problems identified during the outage included cracked or broken springs in the main steam isolation valves (MSIVs) and an incorrect undervoltage trip unit setting designed to protect plant equipment from degraded grid voltage. There are eight springs in each of eight MSIVs; a total of four were found to be broken. Examination of the springs determined that the cause of the failure was a manufacturing defect. The broken springs were replaced and the remaining springs were inspected. The MSIVs are air operated valves, using springs to assist in the valve closing. Even with the broken springs, however, the valves would have closed properly had they been needed.

The degraded grid voltage problem involved an undervoltage trip unit setpoint which was set too low to provide protection for some safety-related equipment (principally certain motors which operate valves). In the event of a drop in the voltage on the off-site electrical power grid supplying power to the plant, the undervoltage trip unit would not shift the power supply to the emergency diesel generators before damage could potentially occur. The corrective action included resetting of the undervoltage trip unit setpoint and a related change in the plant's Technical Specifications (license requirements).

As briefly discussed in the original report (i.e., NUREG-0090, Vol. 8, No. 4), one of the events which raised significant concern regarding management deficiencies at Fermi Unit 2, involved a premature criticality during a reactor startup on July 1, 1985. The NRC has completed its investigation and on July 3, 1986, forwarded to the licensee (Detroit Edison Company) an Order Modifying the License Effective Immediately, and a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $300,000 (Ref. B-5). This event and the actions taken are described in considerably more detail in Appendix B of this report as an update to Abnormal Occurrence No. 85-1 ("Premature Criticality During Startup").

Future reports will be made as appropriate.

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86-1 <u>Loss of Power and Water Hammer Event</u>

This abnormal occurrence, which occurred at San Onofre Unit 1 on November 21, 1985, was originally reported in NUREG-0090, Vol. 9, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1986." It is updated as follows.

As discussed in the report of this abnormal occurrence, the licensee's program for addressing issues stemming from the November 21, 1985 incident was originally presented to the NRC Staff on April 8, 1986. Additional information was submitted on May 1 and May 5, 1986. The Staff has completed their technical review of the licensee submittals and prepared technical evaluation and inspection reports.

The licensee and NRC Staff met with the Commissioners on June 12, 1986. The Commissioners reviewed the licensee's and Staff's actions taken to analyze and correct the deficiencies that led to the event. Principal among these actions was the development and implementation of a Material Condition Review Program. The program was designed to define a suitable material standard for systems and components in an older plant and to ensure that the material condition of those items was maintained.

On June 25, 1986, the Region V Administrator visited the site to assess the material condition of San Onofre Unit 1. Although minor deficiencies were identified, the Unit condition was found to be greatly improved from what it had been prior to the November 21, 1985 event.

Based on the evaluations by the NRC Office of Nuclear Reactor Regulation, the Office of Inspection and Enforcement, and the NRC Region V Office, the Staff concluded that the licensee's plant improvements for resolution of the water hammer event were adequate for facility restart subject to certain procedural changes and additional commitments to which the licensee agreed.

On July 13, 1986, the Region V Administrator concurred with the restart of Unit 1. The reactor was taken critical on July 15, 1986, for low power physics testing, and was connected to the grid on July 26, 1986.

NRC Inspection and Enforcement Information Notice No. 86-49 was issued on June 16, 1986, to all nuclear power reactor facilities holding an operating license or a construction permit (Ref. B-12). The Notice described age/environment failures of electrical cables at San Onofre Unit 1 (which contributed to the November 21, 1985 event). The Notice also suggested actions which licensees could take to improve in-service cable reliability.

On September 17, 1986, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $180,000 (Ref. B-13).

This item is considered closed for the purposes of this report.

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86-2 Loss of Integrated Control System Power and Overcooling Transient

This abnormal occurrence, which occurred at Rancho Seco on December 26, 1985, was originally reported in NUREG-0090, Vol. 9, No. 1, "Report to Congress on Abnormal Occurrences, January-March 1986." It is updated as follows.

On June 12, 1986, Sacramento Municipal Utility District (the licensee) met with the NRC Staff to discuss the scope and content of the preliminary action plan for performance improvement. The intent was to obtain NRC comments so that they could be incorporated into the final report. As a result of the meeting, an NRC team visited the Rancho Seco site on June 17-18, 1986, to review in detail the method being utilized by the licensee to define its system review and test program. The team found that the licensee had a structured program underway to find and correct problems with hardware, procedures, maintenance, etc. However, work on defining the system review and test program had just started and little information was available on program scope or content.

The licensee submitted its action plan for performance improvement on July 3, 1986, and the Staff has initiated its review. On August 18 and September 4, 1986, the licensee met with the NRC Staff to discuss the scope and content of the action plan. The critical path to restart decision is the Staff's review of the action plan and issuance of the Staff's safety evaluation report.

Future reports will be made as appropriate.

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FUEL CYCLE FACILITIES

86-3 Rupture of Uranium Hexafluoride Cylinder and Release of Gases

This abnormal occurrence, involving Sequoyah Fuels Corporation, Gore, Oklahoma, was originally reported in NUREG-0090, Vol. 9, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1986." It is updated as follows.

By letter dated May 8, 1986, Sequoyah Fuels Corporation requested authorization to restart production of UF_6. However, this request was not accompanied by commitments that could be incorporated into the license. Accordingly, by letter dated May 23, 1986, the NRC Staff requested that additional information be provided. A response to this request was received by letter dated June 28, 1986, and the review of the information is complete. Further changes to the license were submitted on August 21, 1986. These documents are under review. NRC has also held a public meeting in Gore, Oklahoma, on July 8-9, 1986, to solicit the concerns of local individuals so that these concerns can be taken into consideration during the restart review. The NRC Staff estimates that action will be taken on the restart request during the latter part of the third calendar quarter of 1986.

As discussed in NUREG-0090, Vol. 9, No. 1, a Lessons-Learned Group was reviewing the event as well as regulatory practices in general regarding such fuel facilities. This Group was formed on February 20, 1986, by the NRC Acting Executive Director for Operations to prepare a report based on experience gained from this event. The goal of the Group was to identify actions NRC might reasonably take from a licensing and inspection standpoint to prevent similar incidents, as well as to clarify NRC's regulatory role regarding facilities of this type. A further goal was to assess the adequacy of the NRC response to the incident, as well as the follow-on activities.

The Group's report was issued during June 1986 as NUREG-1198 (Ref. B-14). The report presents discussions and recommendations on the process, operation and design of the facility, as well as on the responses of the licensee, NRC, and other local, state and federal agencies to the incident. It also provides recommendations in the areas of NRC licensing and inspection of fuel facility and certain other non-power reactor licensees. The implementation of some recommendations will depend on decisions to be made regarding the scope of NRC responsibilities with respect to those aspects of the design and operation of such facilities that are not directly related to radiological safety.

NRC Staff responses to those recommendations were published in August, 1986, as NUREG-1198 Supplement 1 (Ref. B-15). On July 28, 1986, the NRC staff performed the final facility inspection needed for restart authorization, to verify that all license-required changes/improvements had been made. Determination of specific enforcement actions has yet to be completed.

Future reports will be made as appropriate.

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OTHER NRC LICENSEES

84-13 Contaminated Radiopharmaceuticals Used in Diagnostic Administrations

This abnormal occurrence, which occurred at two nuclear pharmacies (i.e., Nuclear Pharmacy, Inc. of Chicago, Illinois, and Syncor International, Inc., of Blue Ash, Ohio), was reported and closed out in NUREG-0090, Vol. 7, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1984." The item is being reopened to provide the following new information.

As discussed in the previous report, on May 18, 1984, the two nuclear pharmacies received faulty devices from Medi-Physics, Inc. (located in Tuxedo, New York) used for preparing doses of technetium-99m, a radioactive pharmaceutical widely used for diagnostic medical tests. The radiopharmaceutical produced by the faulty devices was contaminated with molybdenum-99, another radioactive material. Contrary to NRC requirements, the contaminated radiopharmaceutical was distributed to medical customers of the two nuclear pharmacies and administered to about 21 patients. As a result, these patients received higher internal radiation exposures than necessary.

Also as previously discussed, the NRC completed enforcement action against Syncor by imposing a civil penalty of $8,500 on January 2, 1985 (Ref. B-16). The problems at Nuclear Pharmacy, Inc., were more complex and required more significant enforcement actions. On October 26, 1984, the NRC issued an Order Modifying Nuclear Pharmacy's license (Ref. B-17). This Order imposed additional specific requirements that were to be implemented by November 2, 1984. During the period November 16-19, 1984, the NRC Region III staff conducted an unannounced inspection at the Chicago facility to determine the status of compliance with the October 26, 1984 Order. This inspection disclosed numerous violations of the Order, including many examples of falsification of records. An enforcement conference was held on October 2, 1984, and an investigation was initiated by the NRC Office of Investigations.

On April 10, 1986, the NRC forwarded to Nuclear Pharmacy, Inc., a Notice of Violations and Proposed Imposition of Civil Penalties in the amount of $68,000 (Ref. B-18). The violations were associated with the distribution of the contaminated radiopharmaceutical, including material false statement to the NRC regarding distribution of the radiopharmaceutical and falsification of records indicating that contamination tests had been performed when, in fact, they had not been. In addition, the fine included violations of the October 26, 1984 Order requiring dual verification of certain activities associated with distribution of radiopharmaceuticals, and violations which occurred at several other Nuclear Pharmacy, Inc., facilities. The licensee paid the civil penalty.

In July 1985 Nuclear Pharmacy, Inc., merged with Syncor. Since the merger, the principal individuals involved in Nuclear Pharmacy's May 1984 incident have resigned, been terminated, or are no longer involved in NRC-licensed activities. A new management structure is in place for the merged concern.

This item is considered closed for the purposes of this report.

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86-6 Breakdown of Management Controls at an Irradiator Facility

This abnormal occurrence, which involved Radiation Technology, Incorporated (RTI), Rockaway, New Jersey, was originally reported in NUREG-0090, Vol. 9, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1986." It is updated as follows.

On June 23, 1986, the license was again suspended, effective immediately, based on the findings of an NRC investigation that management directed the bypass of interlocks and safety features and that management had provided false information to the NRC. The suspension was effective pending review of the licensee's request to renew the license. On August 22, 1986, the license was renewed and the suspension lifted following extensive changes in licensee management and procedures. An augmented inspection program is in place and the license will expire in February 1987, requiring a second renewal application and review.

Future reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

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

1. Reactor Vessel Indications at Oconee Unit 1

During March 1986, Duke Power Company (the licensee) notified the NRC that unacceptable indications had been found during an inservice inspection (ISI) at Oconee Unit 1. The plant is a Babcock & Wilcox (B&W)-designed pressurized water reactor located in Oconee County, South Carolina. The pressure vessel was manufactured by B&W.

During the ISI, the reactor vessel to flange shell weld was ultrasonically examined from the flange face. The centerline of the weld is about 32 inches from the examination surface. Twenty-two indications were recorded, all from the clad side of the flange. Only one of the indications extended onto the unclad side of the flange. The indications were located from 24 to 32 inches from the flange. The majority of the indications were located in the base material above the flange weld. Previous examinations conducted from the vessel inner and outer diameter surfaces as well as the unclad side of the flange face did not detect these same indications. The licensee stated that with the exception of one geometric reflector recorded in 1979, there were no recordable indications on this weld. The 1986 examination, however, was the first examination conducted from the clad side of the flange face.

Essentially all of the examination data were obtained with a single 1-inch diameter transducer. Based on the reported size, number and ultrasonic amplitude of the flaw indications in the reactor vessel, additional inspections with transducers of different sizes, angles and frequencies and different modes of ultrasonic wave propagation would have provided supporting information to further confirm and characterize the flaw indications. Duke Power Company, however, elected to complete the ISI with the available examination data, install the vessel closure head and perform a fracture mechanics evaluation of the flaw indications using the measured dimensions.

The licensee evaluated the data as though the indications represented real flaws since there was no evidence to prove otherwise. Nine of these indications were acceptable when compared to the ASME Section XI acceptance standard IWB-3510. The remaining 13 indications were considered acceptable by the licensee by analytical methods permitted by IWB-3600 in accordance with Appendix A (Section XI) procedures.

NRC Region II requested that the NRC Office of Nuclear Reactor Regulation (NRR) review the licensee's fracture mechanics evaluation because the licensee decided to resolve the issue by this analytical method, i.e., Linear Elastic Fracture Mechanics Analysis.

Two meetings were held with the licensee in the NRC Region II office on April 8 and April 21, 1986 during which the licensee presented their inspection results, fracture mechanics evaluation and their proposed action plan. At the April 8, 1986 meeting, the licensee informed the NRC Staff that some of the reflectors may be geometric in origin. This preliminary conclusion was based on a limited qualitative laboratory test on a reactor vessel mock-up at the Babcock & Wilcox, Mt. Vernon, Indiana facility. However, the licensee intended to consider the flaw indications as actual flaws and to complete the fracture mechanics evaluation based on the requirements of ASME Section XI Subarticle IWB-3600.

The Staff reviewed the licensee's submittals, along with additional information requested during the April 21, 1986 meeting, and concluded that Oconee Unit 1 may restart and operate for eight weeks on the condition that the licensee submit, before June 16, 1986, results on the following work effort:

- Perform a comparison of Units 1 and 3 vessel flange geometry, material(s) and cladding and determine the reason that the Unit 3 ISI examination of the same weld did not produce similar results,

- Conduct an ultrasonic examination study on the Mt. Vernon mock-up reactor vessel, and

- Review the original weld design and fabrication history including nondestructive examination records of the weld in question.

In addition, the Staff stated that the NRC reserves the right to re-evaluate its position on the above, in the event that a significant transient occurred on the Unit 1 vessel.

In a letter dated April 24, 1986, the licensee submitted their fracture mechanics evaluation and a summary of the ultrasonic testing results which located and sized the flaw indications.

NRR has concluded that Oconee Unit 1 can be safely returned to full power and operated with actual flaws of the size and circumferential locations described in the safety evaluation report (SER). This conclusion was based on the Staff's review of the fracture mechanics analysis evaluation performed by the licensee, the pressure-temperature limits, and Pressurizer Code Safety Valve Set Point contained in the Oconee Unit 1 Technical Specifications. This conclusion was supplemented with the following conditions:

- The licensee will submit prior to June 16, 1986, a technical report summarizing their on-going ultrasonic testing program on the Mt. Vernon mock-up vessel,

- The Staff will review and determine whether the conclusion that the subject flaw indications are enveloped by the dimensions measured by the licensee is still conservative,

- Because the Staff considers several of the flaws as conditionally acceptable per IWB-3122.4, augmented ISIs based on 10 CFR §50.55.a(g)(4) will be required, and

- At least six months before the next scheduled refueling outage, the licensee will provide a report describing detailed plans for the above augmented ISIs.

On June 13, 1986 the licensee submitted to NRR a report with the requested information. This report stated that from the preliminary results (from the data obtained at Mt. Vernon), it appeared highly probable that the data recorded during the Oconee Unit 1 examination indicated geometric conditions. The report provided a detailed description of the examination of the Mt. Vernon vessel mock-up using the actual calibration block and similar examination equipment to that used during the Oconee Unit 1 examination, and results from those examinations. Specific recommendations for conducting ultrasonic examination(s) from the flange face of the reactor vessel in future ISIs were provided. These techniques were used during the Oconee Unit 2 refueling outage that began in August 1986. The licensee's report is currently undergoing review by the Staff.

The licensee has returned the plant to power operation. This event had no impact on the public health or safety; therefore, it is not considered reportable as an abnormal occurrence.

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2. <u>NRC Augmented Inspection Team Sent to Pilgrim</u>

On April 12, 1986, an NRC Augmented Inspection Team (AIT) was sent to the Pilgrim Nuclear Power Plant after events occurring on April 4 and 12, 1986, raised some concern regarding safe operation of the plant. Pilgrim, operated by Boston Edison Company (the licensee), utilizes a General Electric-designed boiling water reactor. The plant is located in Plymouth County, Massachusetts.

On April 4 and 12, 1986, the reactor scrammed from low power during routine reactor shutdowns. Both scrams were caused by unexpected Group I primary containment isolations. In both cases the isolation signal was promptly reset, but the four outboard main steam isolation valves (MSIVs) could not be promptly reopened. As a result, the main condenser was not available as a heat sink during a portion of each event and the high pressure coolant injection (HPCI) system was operated in the test mode to control most of the subsequent reactor cooldown. In regard to the April 4 scram, a routine reactor shutdown had been initiated earlier on the same day after oil leakage was detected in the main turbine control air system. In regard to the April 12 scram, a routine reactor shutdown had been initiated on April 11 after periodic residual heat removal (RHR) system high pressure alarms were received on both April 10 and 11 indicating that the RHR system was being pressurized by reactor coolant leakage past various valves. An Unusual Event was declared because of the RHR valve leakage.

Following the second scram, NRC Region I issued Confirmatory Action Letter (CAL) No. 86-10 to the licensee on April 12, 1986, which required that all affected equipment be maintained in its as found condition (except as necessary to maintain the plant in a safe shutdown condition) in order to preserve any evidence which would be needed to inspect or reconstruct the events (Ref. C-3).

The AIT conducted a special safety inspection from April 12 through April 25, 1986, to review the circumstances associated with the areas of concern, i.e.,

(1) the spurious Group I primary containment isolations, (2) the failure of the MSIVs to promptly reopen after the isolations, and (3) the recurring pressurizations of the RHR system due to valve leakage. Based on the AIT inspection, the team noted that the licensee's problem solving approaches were carefully structured and appeared thorough. In addition, the team drew the following conclusions for the three areas of concern:

- No root causes for the spurious primary containment isolations on April 4 and 12, 1986, were identified during the inspection period, despite considerable licensee effort. The team did not identify any weaknesses in the licensee's problem solving approach.

- The failure of the MSIVs to re-open following the containment isolations on April 4 and 12 was caused by partial or complete mechanical separation of the valve pilot poppets from the MSIV valve stem assemblies. Pilot poppet set screws did not prevent the poppets from unscrewing from the stem assemblies.

- The RHR pressurization events reflect slow leakage (about 0.5 gpm) past a check valve and two motor operated injection valves in the "B" RHR loop. Lack of RHR pressure instrumentation and the lack of periodic tests of the RHR injection check valves inhibit a more thorough diagnosis. No apparent RHR valve failure mechanism was identified as the reason for this leakage.

- The licensee's conduct of the reactor shutdown on April 11 and 12, 1986, was prudent in light of the recurring RHR pressurization events.

The licensee's root cause evaluations were not completed and corrective actions were not finalized during the AIT inspection. NRC review of these actions will be conducted prior to startup from this outage.

The above AIT findings are documented in Inspection Report No. 50-293/86-17 which was forwarded to the licensee on May 16, 1986 (Ref. C-4). The previously mentioned NRC Region I CAL also required that the licensee submit a written evaluation to the NRC of the events noted above prior to restart and to seek Regional Administrator authorization for restart. The plant remains shut down and is not expected to restart until April 1987.

The actual safety consequences of the events were minimal as the isolation system and MSIV failures were in the conservative direction and there was no indication that the potential existed for a sudden overpressurization and failure of the RHR piping. Therefore, based on the information developed to date, the item is considered below the threshold for abnormal occurrence reporting. It is being reported here because of (1) the item may be perceived by the public to be of public health or safety significance, (2) a special NRC AIT was organized and sent to the plant to review the events and licensee actions, (3) the considerable effort expended by the licensee and the NRC in resolving the issues, and (4) the extensive shutdown being experienced by the plant.

In addition, the leakage past valves forming the isolation barrier between the high pressure reactor coolant system and the low pressure piping of the RHR system has been a recurrent problem in BWRs. This problem was addressed in Abnormal Occurrence No. 84-8 in NUREG-0090, Vol. 7, No. 3 ("Report to Congress on Abnormal Occurrences: July September 1984") and updated in Appendix B of NUREG-0090, Vol. 8, No. 2 ("Report to Congress on Abnormal Occurrences: April-June 1985"); an NRC Office for Analysis and Evaluation of Operational Data (AEOD) Case Study (AEOD/C502) issued in September 1985 (Ref. C-5); and Inspection and Enforcement Information Notice Nos. 84-74 and 84-81 issued on September 28, 1984, and November 16, 1984, respectively (Refs. C-6 and C-7).

The February 1986 problem at Pilgrim was included in Inspection and Enforcement Information Notice No. 86-40 which was issued on June 5, 1986 (Ref. C-8).

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3. Construction Problems at Comanche Peak

On May 2, 1986, the NRC Director of Inspection and Enforcement transmitted to Texas Utilities Electric Company (the licensee) two separate Notices of Violation and Proposed Imposition of Civil Penalties in the amount of $250,000 as a result of multiple failures to meet requirements pertaining to the construction and quality assurance program at Comanche Peak Units 1 and 2 (Ref. C-9), and in the amount of $120,000 as the result of failure in three instances to assure quality control inspectors the organizational freedom to identify problems (Ref. C-10). These issues are discussed separately below.

The Comanche Peak Steam Electric Station (CPSES) Units 1 and 2 are Westinghouse-designed pressurized water reactors and are located in Somervell County, Texas.

Construction and Quality Assurance

The NRC has devoted substantial resources to evaluating the adequacy of construction at the CPSES facility. In addition to the routine and special inspections conducted by NRC Region IV, a Construction Appraisal Team inspection was conducted by the Office of Inspection and Enforcement on January 24 - February 4, 1983, and February 14 - March 3, 1983. From April 13 - 18, 1984, a review by the Special Review Team (SRT) was conducted by representatives of NRC Region II. Subsequently, the NRC Technical Review Team (TRT) was assembled which consisted of approximately 50 specialists from NRC headquarters, NRC regional offices, and consultants, to evaluate and resolve technical issues and issues identified as a result of allegations. The results of the review of the issues by the TRT are documented in Safety Evaluation Report (SER) NUREG-0797, Supplements 7, 8, 9, 10, and 11 (Ref. C-11). These issues include failure to ensure that quality control inspectors were properly qualified and certified, ineffective interactions between the various engineering and construction groups, deficiencies in the quality control inspection program, and failure to properly implement the site's corrective action program. In response to these issues, the licensee has issued a "Comanche Peak Response Team Program Plan and Issue-Specific Actions Plan" to describe planned corrective actions (Ref. C-12).

Two separate special inspections were conducted on November 18 - December 18, 1985, and January 1 - March 4, 1986, respectively, by Region IV, concerning the Unit 1 as-built cable tray inspection program (Ref. C-13) and the procurement and installation of electrical containment penetration assemblies (Ref. C-14) furnished by the Bunker Ramo Corporation. These two inspections identified failures to properly inspect the Unit 1 cable trays and deficiencies in the procurement and installation of electrical containment penetration assemblies.

On June 25, 1986, the licensee was granted an extension in responding to the TRT issues and Region IV special inspections (Ref. C-15). The licensee response (Ref. C-16) is currently under review by the NRC.

Organizational Freedom

As a result of numerous allegations of intimidation, harassment, and discrimination, and the relevance of this issue to the contentions in the ongoing operating license hearing, the NRC undertook a comprehensive review and evaluation of the allegations of intimidation, harassment, and discrimination at CPSES. A report prepared by an NRC Comanche Peak Intimidation Panel (Panel) aided by a Study Team of consultants was issued on November 4, 1985 (Ref. C-17). The NRC staff has reviewed the Panel report, the completed Department of Labor discrimination cases regarding CPSES, the NRC Office of Investigations reports, and the licensee's responses regarding intimidation at CPSES. Three incidents were identified as violations by the NRC staff as a result of this review.

On June 2, 1986, the licensee responded to the violations concerning intimidation, harassment and discrimination and requested further NRC staff consideration with regard to two of the violations (Ref. C-18). The response is under review by the NRC staff.

Because the above issues occurred during facility construction and there was no fuel in the reactors, there were no effects on public health or safety. Therefore, the issues are not reportable as an abnormal occurrence. They are being reported here because (1) they may be perceived by the public to be of public health or safety significance, (2) considerable resources were expended in evaluating the issues, and (3) the considerable civil penalties which have been imposed.

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4. Sabotage of Offsite Power Lines to Palo Verde

Palo Verde is a three unit nuclear power station operated by Arizona Public Service Company and located in Maricopa County, Arizona. Each unit is a Combustion Engineering-designed pressurized water reactor.

On May 14, 1986, beginning at approximately 8:59 p.m., MST, offsite power on 3 of 4 transmission lines was lost within minutes of each other due to phase-to-ground faults on each line. The power remained faulted through the evening until the afternoon of May 15, 1986. During the morning hours of May 15, the licensee found that a rope or strap had been slung over the transmission lines causing a phase-to-ground fault. The fault location on each line was about 35 miles from the plant.

The transmission lines converge on the site from four different directions. The ground straps were far enough apart to require a coordinated effort by several individuals to accomplish the power loss within a few minutes. Unit 1 was in Mode 5 (cold shutdown) for a maintenance outage. Unit 2 remained in Mode 3 (hot standby) pending completion of the investigation of the event. Unit 3 was in the preoperational test phase, with no fuel in the reactor.

The licensee notified local law enforcement agencies and the FBI.

In accordance with Title 18, Section 1365, US Code, the FBI (Phoenix Office) assumed principal jurisdiction of this incident. They coordinated their investigative efforts with the Maricopa County Sheriff's Office, the Arizona Department of Public Safety and NRC Office of Investigations (Region V). The FBI continues to analyze physical evidence obtained at each of the three areas.

The licensee has offered a $25,000 reward for "information leading to the arrest and conviction of the person or persons responsible for the May 14 sabotage of three major high-voltage lines tied to the Palo Verde Nuclear Generating Station."

The event is of considerable concern because it was deliberate, coordinated sabotage by a group of people. However, there was no challenge to plant safety systems because offsite power remained available through the non-faulted transmission line. The plants were all designed to safely shut down without any offsite power. However, until the saboteurs are apprehended, the potential remains for future challenges to the plant safety systems.

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5. NRC Augmented Inspection Team Sent to Palisades

On May 22, 1986, an NRC Augmented Inspection Team (AIT) was sent to the Palisades facility because of NRC concerns regarding continuing equipment problems, as exemplified by multiple failures which occurred during a May 19, 1986 reactor trip. The Palisades facility is a Combustion Engineering-designed pressured water reactor operated by Consumers Power Company. The plant is located in Van Buren County, Michigan.

In the May 19, 1986 incident, the reactor tripped from high pressure after a loss of turbine control power resulted in the closure of the turbine governor and control valves. Although the plant was shut down successfully and the safety of the plant was not jeopardized, several items of equipment did not operate as designed. The turbine bypass valve did not automatically open; one atmospheric dump valve did not open; a letdown intermediate pressure control valve failed (causing the Chemical and Volume Control System relief valve to lift); a primary coolant charging pump did not start; a rod bottom light did not light; and a pressurizer spray valve did not indicate closed.

These equipment problems are representative of a history of multiple equipment failures at Palisades. The equipment failures are of concern due to the potential for serious challenges to the plant's safety systems and due to the heavy burden they place on the reactor operators to maintain the plant in a safe operating condition.

In the most recent Systematic Assessment of Licensee Performance (SALP) report on Palisades, covering the period November 1, 1984, through October 21, 1985, the rating in three areas declined from a Category 2 in the previous rating period to a Category 3 (Ref. C-19). These areas were Maintenance, Surveillance, and Quality Programs and Administrative Controls. These low ratings were, in part, due to a lack of aggressive corrective action by the licensee and poor management controls and attitudes. The ratings were indicative of the NRC's concerns about the licensee's maintenance program and reliability of plant operating and safety equipment. Concerns regarding maintenance were also expressed in an October 30, 1985 NRC Region III Confirmatory Action Letter requiring the licensee to reduce the backlog of maintenance activities (Ref. C-20).

A special task force review was performed by NRC Region III during the period January 10 through March 7, 1986, to examine the operating history of Palisades from 1983 through 1985. The report of this review, issued to the licensee on May 16, 1986 (Ref. C-21), identified general weaknesses with respect to recurrent equipment problems which required significant operation attention; an abnormally high rate of personnel errors; problems with adequacy of procedures and adherence to procedures; and corrective actions which did not resolve recurring problems and were not sufficiently broad to address generic aspects of equipment problems.

On July 1, 1986, the NRC proposed a $50,000 fine against the licensee (Ref. C-22) for violations not associated with the May 19, 1986 reactor trip. The proposed fine was for inadequate testing of valves on two occasions in 1985. The valves were part of the post-accident sampling system which is used for collecting samples from the reactor cooling system. Later testing in December 1985 showed that leakage past the valves was approximately 20 percent over the NRC leakage limit.

Because of the plant's history of equipment and maintenance problems, after the multiple failures associated with the May 19, 1986 reactor trip, NRC Region III issued a Confirmatory Action Letter on May 21, 1986, confirming that the plant would remain shut down for a review of the equipment problems by both the licensee and by an NRC AIT (Ref. C-23).

The various NRC reviews of activities at the Palisades plant suggest that the history of equipment and maintenance problems at the plant are attributable to inadequate maintenance and corrective actions, to inadequate management controls, and to the added burdens placed on the operating staff to maintain the plant in a safe operating condition.

In conformance with the May 21, 1986 Confirmatory Action Letter, the licensee has kept the Palisades plant shut down since May 19, 1986, for a thorough review of plant equipment reliability and upgrade of the maintenance program. The inspection, testing, and maintenance activities remained underway at the time of preparation of this report.

The AIT conducted a special onsite investigation review on May 22-25, 1986. The report of the AIT findings (Ref. C-24) was issued to the licensee on June 30, 1986, as Inspection Report No. 50-255/86017(DRP). The AIT investigation proceeded simultaneously along two paths - equipment reviews and operations department interviews. Although at the termination of the AIT onsite review, root cause determinations for several equipment failures were still in progress, the following significant facts were ascertained:

- Significant weaknesses exist in three aspects of the maintenance function - diagnostics, repair, and post maintenance testing.
- Plant operators have serious concerns regarding the adequacy of maintenance activities and equipment reliability.

- Equipment failures and degraded equipment have placed varying levels of additional burden on the plant operators. With regard to the May 19 trip, this burden did distract operators, but did not significantly jeopardize plant safety.

- The performance of plant operators and the operation of other major or safety-related plant systems were as expected and designed considering the equipment failures that occurred.

- There is a lack of communication and coordination for a maintenance activity from the Work Request stage through verification testing and acceptance for operation.

The actual safety consequences of the equipment failures were minimal; the failures primarily involved non-safety-related systems. No violations, and consequently no enforcement actions, developed as a result of the AIT report findings. Therefore, the item is considered below the threshold for abnormal occurrence reporting. It is being reported here because of (1) the item may be perceived by the public to be of public health or safety significance, and (2) a special NRC AIT was organized and sent to the plant to review the issue and licensee actions.

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6. Fire in Charcoal Filter Tanks at Perry Unit 1

On June 20, 1986, Cleveland Electric Illuminating Company determined that a fire existed in the off-gas treatment system charcoal adsorber filtering units at Perry Unit 1. The plant was shut down at the time of the fire. The licensee received a low power license for the plant on March 18, 1986 and was conducting preoperational tests prior to the fire. Perry Unit 1 is a General Electric-designed boiling water reactor and is located in Lake County, Ohio.

The licensee was testing the cooling system for the rooms containing the charcoal adsorber units. Large industrial space heaters had been placed in the rooms for the test of the cooling capacity of the cooling system. The positioning of one or more of the space heaters may have resulted in excessive temperatures leading to charcoal ignition.

The off-gas system is used to process gases, both non-radioactive and radioactive, which remain after steam is condensed to water in the plant's main condenser. The system includes two parallel subsystems each with four large tanks containing treated charcoal. The charcoal removes radioactive iodine through adsorption, and the system also holds up other radioactive gases for decay prior to release from the plant.

The charcoal adsorption system uses a series of four tanks in each of the subsystems; each tank is about 25 feet in height and 4 feet in diameter and contains 3 tons of charcoal. The fire was in two tanks - the third in each series of four tanks.

The fire was extinguished by adding a nitrogen atmosphere to the tanks; the fire was determined to be out on June 23, 1986. Subsequently, the fires rekindled in both tanks on July 6, 1986, when the licensee began to replace the nitrogen with air as part of planned testing activities. The nitrogen atmosphere was restored and the fire was again extinguished.

The licensee replaced the charcoal in all eight tanks. The tanks where the burning occurred were examined and determined to be undamaged.

There was no health or safety issues associated with the charcoal fires and no releases of any radioactive material to the environment. The event, however, generated significant news media interest.

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7. Water Level Instrumentation Problem at LaCrosse

On June 22, 1986, the water level instrumentation at the LaCrosse nuclear plant gave erratic readings following a reactor trip. Subsequent investigation determined that the actual reactor water level had remained normal, but that the instrument readings, in some instances, were faulty. LaCrosse is an Allis-Chalmers-designed boiling water reactor operated by Dairyland Power Cooperative. The plant is located in Monroe County, Wisconsin.

The reactor trip was caused by an incorrect signal that a main steam isolation valve (MSIV) was not fully open. The signal resulted from a failed relay in the MSIV circuitry. The relay was later replaced.

Initially, all systems functioned normally, but when the MSIVs closed, as designed, and when reactor pressure reached 1000 pounds per square inch, some erratic reactor vessel water level indications occurred:

a. The wide range water level monitor went full scale high. The wide range monitor failure was later found to be caused by the parting of a sensing line. This line was isolated and repaired.

b. One of the three narow range water level monitors dropped to a reading of negative 20 inches, while the other two remained at positive 18 to 20 inches, which is the normal level following a reactor trip. The zero point for these indicators is 3 feet above the top of the core. The low level signal on the one monitor initiated the high pressure core spray system and started the emergency diesel generator. The negative 20 inch level indication then returned to normal levels (agreeing with the other two indicators) and the levels on all three increased to reflect the water added by the core spray system. Core spray was terminated after about 5 minutes of operations.

c. The wide range monitor remained in service and after the broken sensing line was isolated, the monitor indicated a water level of 2 inches above the core. The licensee determined the monitor was out of calibration, apparently caused by the pressure surge when the sensing line parted. After recalibration, the wide range monitor agreed with the level shown on the three narrow range monitors.

d. In addition, during the time when the conflicting readings were observed, an operator entered containment and checked a direct reading reactor water level instrument (a feature unique to the LaCrosse plant). Its level measurement confirmed that there was sufficient water in the core.

The sensing line which parted was repaired, and the water level instrumentation recalibrated before the unit was returned to operation.

Since investigation showed that actual reactor water level had remained normal during the event, there were no effects on public health or safety. It is reported here because of the emphasis associated with maintaining adequate cooling capacity in nuclear power plants.

REFERENCES
FOR APPENDICES

B-1 Case Study Report, "Decay Heat Removal Problems at U.S. Pressurized Water Reactors," AEOD/C503, prepared by the NRC Office for Analysis and Evaluation of Operational Data, December 1985.*

B-2 Letter from A. E. Chaffee, Chief, Reactor Projects Branch, NRC Region V, to Kenneth P. Baskin, Vice President, Nuclear Engineering Safety and Licensing, Southern California Edison Company, forwarding (1) a Notice of Violation and (2) Inspection Report Nos. 50-361/86-11 and 50-362/86-11, Docket Nos. 50-361 and 50-362, June 20, 1986.*

B-3 Confirmatory Action Letter from James G. Keppler, Regional Administrator, NRC Region III, to Wayne H. Jens, Vice President-Nuclear Operations, Detroit Edison Company, Docket No. 50-341, July 16, 1985.*

B-4 Letter from Charles E. Norelius, Director, Division of Reactor Projects, NRC Region III, to Wayne H. Jens, Vice President-Nuclear Operations, Detroit Edison Company, forwarding Inspection Report No. 50-341/85040 (DRP), Docket No. 50-341, January 7, 1986.*

B-5 Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to Walter J. McCarthy, Jr., Chairman of the Board and Chief Executive Officer, Detroit Edison Company, forwarding an Order Modifying the License Effective Immediately and a Notice of Violation and Proposed Imposition of Civil Penalty, Docket No. 50-341, July 3, 1986.*

B-6 Letter from A. E. Chaffee, Chief, Reactor Operations Branch, NRC Region V, to Kenneth P. Baskin, Vice President, Nuclear Engineering Safety and Licensing, Southern California Edison Company, forwarding a Notice of Violation and Inspection Reports Nos. 50-361/86-11 and 50-362/86-11, Docket Nos. 50-361 and 50-362, June 20, 1986.*

B-7 U.S. Nuclear Regulatory Commission, "Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985," USNRC Report NUREG-1154, published July 1985.**

B-8 10 CFR § 50.54(f) letter from Harold R. Denton, Director, NRC Office of Nuclear Reactor Regulation, to Joe Williams, Jr., Senior Vice President, Nuclear, Toledo Edison Company, Docket No. 50-346, August 14, 1985.*

B-9 Letter from John P. Williamson, Chairman and Chief Executive Officer, Toledo Edison Company, to Harold R. Denton, Director, NRC Office of Nuclear Reactor Regulation, Docket No. 50-346, September 10, 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).

**Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555, for inspection. Available for purchase from the GPO Sales Program, Superindendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082.

B-10 U.S. Nuclear Regulatory Commission "Safety Evaluation Report related to the restart of Davis-Besse Nuclear Power Station, Unit 1, following the event of June 9, 1985," USNRC Report NUREG-1177, published June 1986.**

B-11 U.S. Nuclear Regulatory Commission, "Report of the Independent Ad Hoc Group for the Davis-Besse Incident," USNRC Report NUREG-1201, published June 1986.**

B-12 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 86-49, "Age/Environment Induced Electrical Cable Failures," June 16, 1986.*

B-13 Letter from John B. Martin, Regional Administrator, NRC Region V, to D. J. Fogarty, Executive Vice President, Southern California Edison Company, forwarding a Notice of Violation and Proposed Imposition of Civil Penalties, Docket No. 50-206, September 17, 1986.*

B-14 U.S. Nuclear Regulatory Commission, "Release of UF_6 from a Ruptured Model 48Y Cylinder at Sequoyah Fuels Corporation Facility: Lessons - Learned Report," USNRC Report NUREG-1198, published June 1986.**

B-15 U.S. Nuclear Regulatory Commission, "Release of UF_6 from a Ruptured Model 48Y Cylinder at Sequoyah Fuels Corporation Facility: Lessons-Learned Report," USNRC Report NUREG-1198, Supplement 1, published August 1986.**

B-16 Letter from James M. Taylor, Deputy Director, NRC Office of Inspection and Enforcement, to Mark T. Hebner, President and Chief Executive Officer, Syncor International Corporation, forwarding an Order Imposing a Civil Monetary Penalty, License No. 34-18309-01MD, January 2, 1985.*

B-17 Letter from James M. Taylor, Deputy Director, NRC Office of Inspection and Enforcement, to Nuclear Pharmacy, Inc., forwarding an Order Modifying Licenses (effective immediately), License Nos. 12-18044-01MD, 14-19990-01MD, 20-21227-01MD, 37-18461-01MD, 37-19586-01MD, 37-21322-01, and 48-17466-01MD, October 26, 1984.*

B-18 Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to Monty Fu, Chairman of the Board and Chief Executive Officer, Nuclear Pharmacy, Inc., forwarding a Notice of Violation and Proposed Imposition of Civil Penalties, License Nos. 37-21322-01, 12-18044-01MD, 14-19990-01MD, 20-21227-01MD, 37-18461-01MD, 37-19586-01MD, and 48-17466-01MD, April 10, 1986.*

C-1 Confirmatory Action Letter No. 86-10 from Thomas E. Murley, Regional Administrator, NRC Region I, to William D. Harrington, Senior Vice President, Nuclear, Boston Edison Company M/C Nuclear, Docket No. 50-293, April 12, 1986.*

C-2 Letter from Richard Starostecki, Director, Division of Reactor Projects, NRC Region I, to William D. Harrington, Senior Vice President, Nuclear, Boston Edison Company M/C Nuclear, forwarding Augmented Incident Response Team Report (Inspection Report No. 50-293/86-17), Docket No. 50-293, May 16, 1986.*

C-3 U.S. Nuclear Regulatory Commission, Case Study Report, "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors," prepared by NRC Office for Analysis and Evaluation of Operational Data. Preliminary report, dated February 1985 was issued March 7, 1985. Final report (designated as AEOD/C502) was issued in September 1985.*

C-4 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 84-74, "Isolation of Reactor Coolant System from Low-Pressure Systems Outside Containment," September 28, 1984.*

C-5 U.S. Nuclear REgulatory Commission, Inspection and Enforcement Information Notice No. 84-81, "Inadvertent Reduction in Primary Coolant Inventory in Boiling Water Reactors During Shutdown and Startup," November 16, 1984.*

C-6 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 86-40, "Degraded Ability to Isolate the Reactor Coolant System from Low-Pressure Coolant Systems in BWRs," June 5, 1986.*

C-7 Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to W. G. Counsil, Executive Vice President, Texas Utilities Electric Company, forwarding two Notices of Violation and Proposed Imposition of Civil Penalties, Enforcement Action (EA) 86-09, Docket Nos. 50-445 and 50-446, May 2, 1986.*

C-8 Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to William G. Counsil, Executive Vice President, Texas Utilities Electric Company, forwarding a Notice of Violation and Proposed Imposition of Civil Penalties, Enforcement Action (EA) 86-63, Docket Nos. 50-445 and 50-446, May 2, 1986.*

C-9 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station Units 1 and 2," Docket Nos. 50-445 and 50-446, USNRC Report NUREG-0797, published July 1981.** Pertinent Supplements published as follows:

Supplement No. 7 published January 1985.**
Supplement No. 8 published February 1985.**
Supplement No. 9 published March 1985.**
Supplement No. 10 published April 1985.**
Supplement No. 11 published May 1985.**

C-10 Letter from W. G. Counsil, Executive Vice President, Texas Utilities Generating Company, to Vincent S. Noonan, Director, Comanche Peak Project, NRC Office of Nuclear Reactor Regulation, forwarding a report ("Comanche Peak Steam Electric Station, Units 1 and 2, Comanche Peak Response Team Program Plan and Issue-Specific Action Plans, Revision 3"), Docket Nos. 50-445 and 50-446, January 27, 1986.*

C-11 Letter from E. H. Johnson, Director, Division of Reactor Safety and Projects, NRC Region IV, to W. G. Counsil, Executive Vice President, Texas Utilities Generating Company, forwarding Inspection Report No. 50-445/85-19, Docket No. 50-445, March 28, 1986.*

C-12 Letter from E. H. Johnson, Director, Division of Reactor Safety and Projects, NRC Region IV, to W. G. Counsil, Executive Vice President, Texas Utilities Generating Company, forwarding Inspection Report Nos. 50-445/86-04 and 50-446/86-03, Docket Nos. 50-445 and 50-446, March 27, 1986.*

C-13 Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to W. G. Counsil, Executive Vice President, Texas Utilities Electric Company, Docket Nos. 50-445 and 50-446, June 25, 1986.*

C-14 Letter from W. G. Counsil, Executive Vice President, Texas Utilities Generating Company, to James M. Taylor, Director, NRC Office of Inspection and Enforcement, Docket Nos. 50-445 and 50-446, August 4, 1986.*

C-15 Letter from Vincent S. Noonan, Director, Comanche Peak Project, NRC Office of Nuclear Reactor Regulation, to W. G. Counsil, Executive Vice President, Texas Utilities Generating Company, forwarding a report ("Report of the Review and Evaluation of Allegations of Intimidation and Harassment of Employees at Comanche Peak Steam Electric Station Units 1 and 2," October 1985), Docket Nos. 50-445 and 50-446, November 4, 1985.*

C-16 Letter from W. G. Counsil, Executive Vice President, Texas Utilities Generating Company, to James M. Taylor, Director, NRC Office of Inspection and Enforcement, Docket Nos. 50-445 and 50-446, June 2, 1986.*

C-17 Letter from James G. Keppler, Regional Administrator, NRC Region III, to Dr. F. W. Buckman, Vice President, Nuclear Operations, Consumers Power Company, enclosing the "Systematic Assessment of Licensee Performance Report for the Palisades Nuclear Generating Station," Docket No. 50-255, February 12, 1986.*

C-18 Confirmatory Action Letter (CAL)-RIII-85-15 from James G. Keppler, Regional Administrator, NRC Region III, to R. B. DeWitt, Vice President, Nuclear Operations, Consumers Power Company, Docket No. 50-255, October 30, 1985.*

C-19 Letter from James G. Keppler, Regional Administrator, NRC Region III, to Dr. F. W. Buckman, Vice President, Nuclear Operations, Consumers Power Company, enclosing Task Force Report dated May 1, 1986, for Palisades Nuclear Generating Station, Docket No. 50-255, May 16, 1986.*

C-20 Letter from James G. Keppler, Regional Administrator, NRC Region III, to Dr. F. W. Buckman, Vice President, Nuclear Operations, Consumers Power Company, forwarding (1) a Notice of Violation and Proposed Imposition of Civil Penalty and (2) Inspection Report No. 50-255/86008 (DRP), Docket No. 50-255, July 1, 1986.*

C-21 Confirmatory Action Letter (CAL)-RIII-86-002 from James G. Keppler, Regional Administrator, NRC Region III, to Dr. F. W. Buckman, Vice President, Nuclear Operations, Consumers Power Company, Docket No. 50-255, May 21, 1986.*

C-22 Letter from Charles E. Norelius, Director, Division of Reactor Projects, NRC Region III, to Dr. F. W. Buckman, Vice President, Nuclear Operations, Consumers Power Company, forwarding Augmented Investigation Team Report No. 50-255/86017 (DRP), Docket No. 50-255, June 30, 1986.*

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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 April 1 through June 30, 1986. During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved an out of sequence control rod withdrawal and the other involved a boiling water reactor emergency core cooling system design deficiency. There were five abnormal occurrences at the other NRC licensees. Two involved willful failure to report diagnostic medical misadministrations to the NRC; one involved a therapeutic medical misadministration; and two involved diagnostic medical misadministrations. There were two abnormal occurrences reported by the Agreement States. One involved an uncontrolled release of krypton-85 to an unrestricted area; the other involved a contaminated radiopharmaceutical used in diagnostic administrations. The report also contains information updating some previously reported abnormal occurrences.

14. DOCUMENT ANALYSIS — a. KEYWORDS/DESCRIPTORS

Personnel Errors; Management Deficiency; Control Rod; BWR ECCS Design Deficiency; Diagnostic Medical Misadministration; Therapeutic Medical Misadministration; Uncontrolled Release; Krypton-85; Reactor Vessel Flaw; AIT Inspection; Primary Containment Isolations; Construction

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