NUREG-0090
Vol. 10, No. 4

Report to Congress on Abnormal Occurrences

October - December 1987

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

Available from

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

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

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

NUREG-0090
Vol. 10, No. 4

Report to Congress on Abnormal Occurrences

October - December 1987

Date Published: March 1988

Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Previous Reports in Series

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NUREG-0090, Vol.7, No.4, October-December 1984, published May 1985

NUREG-0090, Vol.8, No.1, January-March 1985, published August 1985

NUREG-0090, Vol.8, No.2, April-June 1985, published November 1985

NUREG-0090, Vol.8, No.3, July-September 1985, published February 1986

NUREG-0090, Vol.8, No.4, October-December 1985, published May 1986

NUREG-0090, Vol.9, No.1, January-March 1986, published September 1986

NUREG-0090, Vol.9, No.2, April-June 1986, published January 1987

NUREG-0090, Vol.9, No.3, July-September 1986, published April 1987

NUREG-0090, Vol.9, No.4, October-December 1986, published July 1987

NUREG-0090, Vol.10, No.1, January-March 1987, published October 1987

NUREG-0090, Vol.10, No.2, April-June 1987, published November 1987

NUREG-0090, Vol.10, No.3, July-September 1987, published March 1988

ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from October 1 to December 31, 1987.

The report states that for this reporting period, there was one abnormal occurrence at the NRC licensees; the item involved the suspension of license of an oil and gas well tracer company for noncompliance with NRC regulatory requirements. There were no abnormal occurrences reported by the Agreement States.

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

CONTENTS

PREFACE

INTRODUCTION

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

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

The NRC has reviewed Licensee Event Reports, licensing and enforcement actions (e.g., notices of violations, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report covers the period from October 1 to December 31, 1987.

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

THE REGULATORY SYSTEM

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

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

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

REPORTABLE OCCURRENCES

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

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

In addition, the NRC and the nuclear power industry have ongoing efforts to improve the operational data system which include not only the type, and quality, of reports required to be submitted, but also the 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 to NPRD system data, it would assume responsibility for management and funding of the NPRD system. INPO reports that significant improvements in licensee participation have been 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 are applied uniformly to events at NRC and Agreement State licensee facilities. Procedures have been developed and implemented and abnormal occurrences reported by the Agreement States to the NRC are included in these quarterly reports to Congress.

FOREIGN INFORMATION

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

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES
OCTOBER - DECEMBER 1987

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the fourth calendar quarter of 1987. As of the date of this report, the NRC had not determined that any events were abnormal occurrences for that period.

* * * * * * * * *

FUEL CYCLE FACILITIES

(Other Than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the fourth calendar quarter of 1987. As of the date of this report, the NRC had not determined that any events were abnormal occurrences for that period.

* * * * * * * * *

OTHER NRC LICENSEES

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

There are currently about 9,000 NRC nuclear material licenses in effect in the United States, principally for use of radioisotopes in the medical, industrial, and academic fields. Incidents were reported in this category from licensees such as radiographers, medical institutions, and byproduct material users.

The NRC is reviewing events reported by these licensees during the fourth calendar quarter of 1987. As of the date of this report, the NRC had determined that the following event was an abnormal occurrence.

87-20 Suspension of License of an Oil and Gas Well Tracer Company

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criteria) of this report notes that major deficiencies in management controls for licensed facilities or material can be considered an abnormal occurrence.

Date and Place - On October 30, 1987, the NRC issued an Order Suspending License (Effective Immediately) and Order to Show Cause why the license should not be revoked to Tracer Profiles, Inc., of Oklahoma City, Oklahoma (Ref. 1).

Nature and Probable Consequences - During an NRC inspection at the company on March 5-6, 1987, several violations of NRC requirements were identified (Ref. 2). Prior to and following an enforcement conference held on March 26, 1987 with the Vice President of the company, the licensee agreed to several specific corrective actions which were documented in Confirmatory Action Letters (CALs) dated March 13 and April 22, 1987 (Refs. 3 and 4, respectively). Among other actions, these included obtaining the services of a qualified consultant to audit operations, develop management controls to ensure compliance with license requirements, and prepare a report of findings which should be forwarded to the NRC.

On June 8, 1987, a Notice of Violation (NOV) (Ref. 5) was issued in which the violations were categorized in the aggregate as a Severity Level III (on a scale in which Severity Levels I and V represent the most and least severe, respectively) without the usual proposed imposition of a civil penalty in consideration of the licensee's past good enforcement history and agreement to implement the corrective actions documented in the CALs.

The licensee failed to respond to the CALs and the NOV. Subsequent attempts to contact licensee management were unsuccessful until July 20, 1987, when the President of the company called the NRC Region IV office and advised that he was unaware of the Vice President's whereabouts and the company's commitments to the NRC and the subsequent NOV. The President consequently committed to additional corrective actions, including securing licensed materials in locked storage until NRC approved resumption of licensed activities. (The licensee apparently possessed only short-lived radionuclides, which had decayed to insignificant levels.) The commitments were formalized in a CAL dated July 31, 1987 (Ref. 6).

However, the NRC did not receive a response. In addition, it has been determined that the company vacated its offices and moved to a new and unknown location without notifying the NRC. Consequently, the NRC issued the previously mentioned Order on October 30, 1987 (Ref. 1).

Cause or Causes - The cause is the licensee's failure to fulfill its commitments to the NRC and its apparent inability and unwillingness to comply with NRC regulatory requirements.

Actions Taken to Prevent Recurrence

Licensee - None.

NRC - The NRC is considering action to revoke the license.

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

* * * * * * * *

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the fourth calendar quarter of 1987, the Agreement States reported no abnormal occurrences to the NRC.

* * * * * * * *

REFERENCES

1. Letter from James M. Taylor, NRC Deputy Executive Director for Regional Operations, to James M. Gibson, President, Tracer Profiles, Inc., forwarding an Order Suspending License (Effective Immediately) and Order to Show Cause, License No. 35-21272-01, Docket No. 30-20132, October 30, 1987.*

2. Letter from William L. Fisher, Chief, Radiological and Safeguards Programs Branch, Division of Radiation Safety and Safeguards, NRC Region IV, to Jack T. Carter, Vice President, Tracer Profiles, Inc., forwarding Inspection Report No. 30-20132/87-01, License No. 35-21272-01, Docket No. 30-20132, March 20, 1987.*

3. Confirmation of Action Letter, from Robert D. Martin, NRC Regional Administrator, NRC Region IV, to Jack T. Carter, Owner, Tracer Profiles, Inc., License No. 35-21272-01, Docket No. 30-20132, March 13, 1987.*

4. Confirmation of Action Letter, from Robert D. Martin, Regional Administrator, NRC Region IV, to Jack T. Carter, Owner, Tracer Profiles, Inc., License No. 35-21272-01, Docket No. 30-20132, April 22, 1987.*

5. Letter from Robert D. Martin, Regional Administrator, Region IV, to Jack T. Carter, Vice President, Tracer Profiles, Inc., forwarding a Notice of Violation, License No. 35-21272-01, Docket No. 30-20132, June 8, 1987.*

6. Confirmation of Action Letter, from Robert D. Martin, Regional Administrator, NRC Region IV, to James M. Gibson, President, Tracer Profiles, Inc., License No. 35-21272-01, Docket No. 30-20132, July 31, 1987.*

---

*Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555, for public inspection and/or copying.

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.

For Commercial Nuclear Power Plants

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

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

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

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

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

For Fuel Cycle Licensees

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

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

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

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the October through December 1987 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. 10, No. 3. It is planned to continue these updates until defueling activities at the site are completed. The update of activities for the period of October 1 through November 30, 1987 (except where otherwise noted) is as follows.

Reactor Building Activities

During the reporting period (October-November 1987), 60 entries were made into the TMI-2 reactor building, bringing the total number of entries since the March 1979 accident to 1470. Reactor building activities during this period centered on the continuing defueling operation, including data acquisition, video inspection, bulk defueling, debris vacuuming and removal of standing, partial, fuel assemblies. Hydrolazer scarifying (removal of a shallow surface layer of concrete with a high pressure water jet) of contaminated surfaces in the reactor building basement was initiated.

Reactor Vessel and Ex-Vessel Defueling Operations

During the reporting period, 29,000 pounds of debris were removed from the reactor vessel, including 35 partial length fuel assemblies. The removal of partial length fuel assemblies accounted for virtually all of the core debris mass removed from the reactor vessel during the period, although smaller items of debris were removed by pick and place processes and by the use of the air lift tool.

At the close of the reporting period, 175 of 177 fuel assemblies had been removed and loaded into defueling canisters. The total mass loaded into canisters was approximately 193,000 lbs (64 percent) out of a total of approximately 300,000 lbs of core debris and other materials. The total mass to be removed includes the mass of the core, structural and absorber materials, mass added by oxidation of core and structural material, and portions of the baffle plates, formers, and other components that will become comingled with core debris during cutting operations. The remaining fuel assembly residuals consist of the remnants of two melted and resolidified fuel assemblies. Their condition made their removal difficult. However, the removal of partial length fuel assemblies was completed in December 1987. This completes defueling of the core region.

The next two regions to be defueled are the lower internals interstices and the lower head region (below the normal core region). Observations with video equipment have revealed two holes in the core former wall adjacent to the two remaining fuel assembly remnants. One hole is about 6 by 4 inches; the other, only partially visible, is about 5 feet high with a width varying between 8 and 30 inches. Two melted core former plates as well as thermal damage to the core barrel are visible through the latter hole. A more definitive video examination was completed in December.

Further testing and checkout of tools to be used in disassembly and defueling of the lower core support assembly (part of the lower internals) was also performed.

Ex-vessel defueling activities concentrated on defueling of the "B" steam generator and the pressurizer. Using long-handled tools and a vacuum, workers removed approximately 25 pounds of debris from the "B" steam generator. Studies underway will determine whether additional defueling is necessary. In addition, about 20 pounds of material were removed from the pressurizer. An additional 200 pounds remain in a hard-packed form. Tooling is being developed to remove this material.

Cask and Liner Shipments

Offsite shipments of TMI-2 core debris to INEL continued during this period; three shipping cask loads, of seven defueling canisters each, were transferred by rail. As of the end of the period, approximately 134,300 lbs of core debris (45 percent of the total estimated quantity) had been shipped. Thirteen EPICOR liners, two high integrity containers of dry activated waste, two liners containing auxiliary building sump sediment, and a submerged demineralizer system Cuno filter were also shipped offsite during the reporting period.

EPICOR II/Submerged Demineralizer System (SDS) Processing

Through the end of the reporting period, a total of 4,505,022 gallons of water had been processed through the SDS and a total of 3,905,827 gallons had been processed through the EPICOR II system. For the reporting period, approximately 208,500 gallons were processed by the EPICOR II system. SDS remained shut down.

Auxiliary and Fuel Handling Building (AFHB) Activities

Decontamination activities continued in the TMI-2 AFHB during the reporting period. These activities centered around steam vacuum cleaning, scabbling, and hands-on decontamination of AFHB cubicles. The robot, Louie-2, continued to be used to scabble areas of the highly contaminated seal injection valve room. Also, several attempts were made, with only limited success, to move resins from the makeup and purification demineralizer vessels to spent resin storage tanks. The removal of these resins will facilitate the decontamination of related systems and cubicles. As of the end of the reporting period, twenty-six cubicles have been decontaminated during 1987.

Post-Defueling Monitored Storage

The NRC is evaluating the licensee's plans for Post-Defueling Monitored Storage and expects to issue a draft environmental statement during the first calendar quarter of 1988.

Proposal to Dispose of Accident-Generated Water

The NRC staff reviewed the licensee's proposal and published on June 30, 1987 the Programmatic Environmental Impact Statement Final Supplement No. 2, NUREG-0683, dealing with disposal of the accident-generated water (Ref. B-1). In Supplement 2, the staff concluded that the licensee's proposal to dispose of the water by forced evaporation to the atmosphere, followed by onsite solidification of the remaining solids and disposal of the solids at a licensed low-level radioactive waste disposal facility, is an acceptable plan. An opportunity for a prior hearing on the staff's proposal to lift the current prohibition on the disposal of the contaminated water was offered to GPUNC and to other persons who may be affected.

On November 3, 1987, the Atomic Safety and Licensing Board issued an order scheduling a pre-hearing conference on December 8, 1987 related to the proceedings on disposition of the accident-generated water. The purposes of the conference, which was open to the public, were to identify the key issues, take any necessary steps for further identification of the issues, make determinations as to the parties to the proceedings, and to establish a schedule for further actions in the proceedings.

TMI-2 Advisory Panel Meeting

The Advisory Panel for the Decontamination of Three Mile Island Unit 2 did not meet during the reporting period.

Future reports will be made as appropriate.

* * * * * * * *

85-6 Significant Deficiencies in Reactor Operator Training and Material False Statements

This abnormal occurrence was originally reported, and closed out, in NUREG-0090, Vol. 8, No. 2 ("Report to Congress on Abnormal Occurrences: April-June 1985"). It is being reopened, and then reclosed, to report new, significant information.

As previously reported, by letter of June 3, 1985, the NRC issued to Mississippi Power and Light Company (MP&L), operator of the Grand Gulf facility at the time, a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $500,000 for alleged deficiencies in the reactor operator training program and alleged material false statements made to the NRC (Ref. B-2). Applications for reactor operator licenses containing apparently false information were submitted to the NRC in September 1981, March 1982, and May 1982. The alleged violations, classified as high as Severity Level I (on a scale where Severity

Levels I and V are the most and least severe, respectively), were documented by special inspections by the NRC Region II Office and investigations by the NRC Office of Investigations. The licensee contested the proposed penalty and their response was reviewed by the NRC staff.

On December 20, 1986, the NRC authorized transfer of control and performance of licensed activities from MP&L to System Energy Resources, Inc. (SERI), formerly named Middle South Energy, Inc. The MP&L personnel and organizations involved in the operation of Grand Gulf were transferred essentially intact to SERI.

On October 22, 1987, SERI signed a Settlement Agreement with the NRC which resolves the Notice of Violation without finally determining the validity of the alleged violations, and mitigates the proposed \$500,000 civil penalty to \$200,000. On October 29, 1987, SERI paid the \$200,000 civil penalty.

In the Agreement, SERI has acknowledged that at the time of the alleged violations there had been serious deficiencies at the Grand Gulf facility in that (1) there was a need to improve operator training procedures and their implementation, (2) errors and misstatements were made in certain operator license applications in that time frame, and (3) better management supervision was necessary to prevent errors and misstatements to the NRC, to ensure accurate, complete and timely information is provided to the NRC, and to ensure that the NRC is informed of incorrect information submitted to it and that prompt action is taken to correct erroneous submissions.

In agreeing to mitigate the civil penalty, the NRC staff has acknowledged that SERI "has taken aggressive and effective corrective actions in the five years since the violations occurred to address deficiencies in the licensed operator training program...." In addition, citing its most recent assessment of licensee performance, the NRC staff observed that the trend in performance at Grand Gulf is improving. The NRC concluded that a settlement would be in the public interest in light of the licensee's acknowledgement of the serious deficiencies involved, the extensive corrective actions undertaken, and the improved performance in the area of training at Grand Gulf.

The NRC believes that, under the special circumstances of the case, the Agreement best serves the interest of both parties and the purpose of the Atomic Energy Act and the NRC's requirements.

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

* * * * * * * *

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; Vol. 9, No. 2; Vol. 9, No. 3; and Vol. 10, No. 2. It is further updated to describe a fire during this report period at one of the Tennessee Valley Authority (TVA) plants which has resulted in extensive investigations by licensee and government personnel.

On November 2, 1987, a fire occurred in the drywell of the Browns Ferry Nuclear Power Station, Unit 2. Browns Ferry Units 1, 2, and 3 are General Electric-designed boiling water reactors located in Limestone County, Alabama. The three Browns Ferry units have been shut down since March 1985 and will remain shut down until numerous significant programmatic and management deficiencies (as described in previous reports in this series) are satisfactorily resolved.

The fire was detected by a contractor quality control inspector who noticed electrical arcing in the cable trays at 10:45 a.m. (E.S.T.). Welding and electrical modification activities were ongoing in the vicinity at the time of the fire. The plant fire brigade responded and extinguished the fire at 11:20 a.m., using both carbon dioxide and water. No outside assistance was required. There were no personnel injuries. The licensee declared an unusual event at 11:08 a.m., which was subsequently cancelled at 12:35 p.m. The NRC received initial notification of the event by telephone from the licensee to the NRC Headquarters Incident Response Center at 11:40 a.m.

The fire burned electrical cables in three cable trays adjacent to the drywell. The following systems were affected by the fire: drywell and floor drain sump pumps, recirculation loop valves, nuclear instrumentation, and drywell blowers and dampers. No equipment damage besides the cables was apparent; however, some aluminum conduits over the upper cable tray were melted due to the intense heat from the fire. The affected electrical cable trays contained permanent and temporary 480, 240, and 120 volt power cables. Since there was no fuel in the reactor (all fuel was in storage in the spent fuel pool), any potential effect on public health or safety was minimal.

The licensee established a special corporate team to investigate the cause of the fire. Four of the sixteen samples taken from fire debris and analyzed for accelerants showed traces of gasoline thereby indicating the fire may be of suspicious origin. The licensee has requested assistance from the Bureau of Alcohol, Tobacco, and Firearms for investigation of the fire. The FBI was also notified. The licensee is continuing to investigate other possible causes of the fire.

The fire area and associated electrical breakers were quarantined by the NRC. The NRC dispatched a special onsite inspection team to evaluate the root cause and follow the licensee investigation. The NRC continues to be involved in the resolution of this event and related matters.

Considerable effort is required to resolve any issues associated with the fire. This may have an impact on the overall schedule for returning the plants to power operation.

Future reports will be made as appropriate.

* * * * * * * *

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," and updated in NUREG-0090, Vol. 9, No. 2; and Vol. 9, No. 3. It is further updated through the end of December 1987 as follows.

Reporting under this abnormal occurrence consists of two parts: (a) progress made toward restart of Rancho Seco, which was brought to a cold shutdown condition following the December 26, 1985 cooldown transient; and (b) progress made in a comprehensive reassessment of Babcock & Wilcox (B&W)-designed plants to achieve safety and performance improvement. The latter is reported as an Annex to abnormal occurrence 86-2.

In regard to the Rancho Seco plant, as of the end of December 1987 the plant remains shut down. The licensee continues to resolve the various issues (which have been previously discussed in these quarterly reports) which must be satisfactorily complete before the plant is permitted to restart. Based on information available as of the end of December 1987, plant restart may occur during the first calendar quarter of 1988.

Future reports will be made as appropriate.

<u>Annex</u>

<u>Reassessment of Babcock and Wilcox (B&W)-Designed Plants</u>

As discussed in the previous reports, the reassessment was initiated because the NRC was concerned that despite improvements since the TMI accident, the number and complexity of events in plants with reactors designed by B&W had not decreased as expected. The events that occurred at Davis-Besse Nuclear Power Station in June 1985 and at Rancho Seco Nuclear Generating Station in December 1985 reinforced the NRC staff's concern. By letter dated January 24, 1986, the NRC Executive Director for Operations (EDO) informed the Chairman of the Babcock & Wilcox Owners Group (BWOG) that a number of events at B&W-designed reactors led the NRC staff to conclude that the basic design requirements for B&W reactors needed to be reexamined.

In recognition of the fundamental responsibility of each B&W plant owner to ensure that its plant(s) is properly designed and safely operated, the NRC staff encouraged the BWOG to assume a leadership role in accomplishing key aspects of the overall effort required for the reassessment of B&W plants.

By letter dated February 13, 1986, the BWOG committed to take the lead in a planned effort to define concerns relative to reducing the frequency of reactor trips and the complexity of post-trip response in B&W plants. The BWOG described its program in BAW-1919, "Safety and Performance Improvement Program." The fifth and final revision to BAW-1919 was submitted on July 22, 1987.

The objectives of the generic evaluation of B&W plants was to reassess the basic design requirements and to reassess the operational characteristics of these plants. Also, the study compared the overall safety of B&W plants with that of other pressurized water reactors. Potential improvements to reduce the frequency of complex post-trip response to anticipated operational occurrences have been identified.

To achieve the objectives, the BWOG and the NRC staff adopted a multifaceted approach. Included as part of the reassessment was (1) review of operational transients that have occurred at B&W plants; (2) feedback from operational and maintenance personnel, along with the views of NRC regional personnel and resident inspectors; (3) deterministic assessments; (4) probabilistic assessments; and (5) computer simulations.

On June 25, 1986, the program for reassessing B&W plants was discussed with the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on B&W Plants. On September 12, 1986, the NRC staff and the BWOG met with the full ACRS committee to further discuss the program. The BWOG discussed its program with the Commission on November 6, 1986; on August 5, 1987, the BWOG and NRC staff briefed the Commissioners on the status of the BWOG program.

The BWOG program as contained in BAW-1919 consists of reviews of the following 11 significant task areas: Sensitivity Study; Operating Experience; Integrated Control System/Non-Nuclear Instrumentation; the Main Feedwater System; the Emergency Feedwater/Auxiliary Feedwater System; the Instrument Air System; Main Steam Pressure Control System; Operator Burden; Risk Assessment; Operations/Maintenance Personnel Interviews; and Emergency Procedures. In addition, as part of the program, in October 1987 the BWOG submitted a report on events that initiate reactor trips, bringing the significant task areas to a total of 12.

In light of BWOG's lead role in the reassessment of B&W plants, a large part of the NRC staff activities involved interaction with the BWOG in working level meetings and review of the results of the BWOG efforts. To a limited degree, the NRC staff also reviewed B&W plant operating experience. Additionally, the NRC staff assessed the areas of human performance and probabilistic risk assessment.

During November 1987, the NRC issued NUREG-1231, "Safety Evaluation Report Related to Babcock & Wilcox Owners Group Plant Reassessment Program" (Ref. B-3). This safety evaluation report included a review of all principal areas of the BWOG program with the exception of the reactor trip initiating events review, the integrated control system/non-nuclear instrumentation system review and the initiation and control system part of the emergency feedwater system review. These areas are currently under review. This review, as well as other evaluations which may be necessary, will be issued as supplements to NUREG-1231.

In NUREG-1231, the NRC staff concluded that the BWOG plant reassessment program was a broad-based and comprehensive program and that the BWOG's lead role, as well as the oversight provided by the NRC staff, provided a balanced approach that made the effort succeed. The NRC staff also concluded that, in general, the recommendations developed by the BWOG were appropriate and when satisfactorily implemented, should lead to a continuing improvement in the post-trip response for B&W plants. The NRC staff specifically noted that the BWOG efforts in one area, human factors, was deficient primarily because human factors expertise was not applied and offered recommendations to the BOWG to correct the deficiencies.

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

* * * * * * * * *

OTHER NRC LICENSEES

86-23 Release of Americium-241 Inside a Waste Storage Building at Wright-Patterson Air Force Base

This abnormal occurrence was originally reported in NUREG-0090, Vol. 9, No. 4, "Report to Congress on Abnormal Occurrences: October-December 1986." It is updated through November 30, 1987 as follows.

The licensee has completed the decontamination of the facility where americium-241 spills occurred on September 18, 1986, and on October 6, 1986. The major decontamination was completed in November 1986, leaving less than 500 microcuries of residual fixed contamination in the structure. Radioactive waste generated during the 1986 decontamination effort was shipped off-site to a licensed low-level radioactive waste burial site at Richland, Washington.

In September 1987 the building was dismantled and prepared for shipment to a waste disposal site at Barnwell, South Carolina. Surveys of the building site have been completed by the licensee and by an NRC contractor, Oak Ridge Associated Universities, and the results are still being evaluated prior to releasing the site for unrestricted use.

During the investigation of the original contamination incident, it was learned that the individuals involved had stopped at a water faucet to wash off before going to the base radiation services office. The water faucet is located in an area which is used periodically as a camping area for Boy Scouts. Surveys by the licensee shortly after the contamination incidents and subsequently by the NRC determined that there was no detectable radioactive contamination of the faucet area or camping area.

The NRC Office of Investigations (OI) conducted an investigation of the circumstances surrounding the contamination incident and the subsequent handling of the event by the licensee.

The investigation determined that the americium-241 involved in the spill was informally transferred without authorization to the Air Force in the 1970s by John C. Haynes of Newark, Ohio. Haynes was licensed by the Atomic Energy Commission to possess and use the americium-241 for research in changing the color of gemstones. (Contamination and subsequent cleanup of Haynes' laboratory was reported as abnormal occurrence 85-4 in NUREG-0090, Vol. 8, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1985.")

The OI report was forwarded to the Department of Justice in October 1987 for further review and possible investigation.

There was significant interest by Ohio news media in the contamination and subsequent investigation. A hearing was held by Ohio's Senators John Glenn and Howard M. Metzenbaum on November 21, 1987, in Dayton, Ohio, to review the circumstances of the incident.

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

* * * * * * * * *

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.

Occasionally, this Appendix may include events involving exposures to very small areas of the skin (one square centimeter or less) which technically exceed the exposures shown in Appendix A (see Example 1 of "For All Licensees") of this report. The radiobiological literature indicates that an overexposure to a small area of skin (less than one square centimeter) would have much less health significance than a similar dose to larger areas of the body; consequently, such exposures would generally not be considered a major reduction in public health or safety (the general abnormal occurrence criterion) and therefore not reportable as abnormal occurrences. However, all such events, together with the circumstances associated with the events, are reviewed individually to determined their relative significance, and if warranted, will be reported as abnormal occurrences.

1. Fire in Turbine Building at Fort St. Vrain

On October 3, 1987, the NRC sent an Augmented Inspection Team (AIT) to the Fort St. Vrain Nuclear Generating Station to investigate a fire which occurred in the turbine building October 2-3, 1987. Fort St. Vrain is the only high temperature gas reactor in the United States. It is operated by the Public Service Company of Colorado near the town of Platteville, Weld County, Colorado.

On October 2, 1987, the reactor plant was in a startup mode at approximately 26 percent power. There was a failure of a thermal relief valve. Hydraulic oil sprayed onto hot parts of steam system safety relief valves and ignited. The fire was found and initially extinguished by use of a dry chemical fire extinguisher. It was not possible to isolate the spraying hydraulic oil locally because of dense smoke. The fire reignited and grew in size. It took several minutes for the hydraulic oil source to be isolated by valves. The hydraulic oil system at Fort St. Vrain is a safety-related system which operates at 3000 psig. It is unique to the high temperature gas reactor design and does not have a counter-part in light water reactors.

After the hydraulic oil to the failed thermal relief valve was secured, the fire brigade quickly extinguished the fire using water (in the form of fog). The local fire department also responded to the fire at the request of the plant staff, but arrived after the fire was extinguished.

The fire created heavy smoke in the turbine building and burned cables resulting in a loss of some indication and control to the plant operators. The reactor was tripped and helium coolant flow was lost for 12 minutes. This loss of flow event was well within the plant design basis (90 minutes). Later calculations indicated that a loss of flow for 18 hours would have not yielded unacceptable results for the power history existing at the time of the fire.

The fire damaged a multipair telephone cable, thus reducing the offsite telephone capability to the Emergency Notification System, dedicated special purpose lines, and other telephone lines. The licensee declared an alert, which remained in effect until the reactor was cooled to less than 200°F. The reactor cooldown was accomplished using normal cooldown methods. The emergency cooldown methods were demonstrated to be operable, but were not used for the cooldown after the fire.

During the fire, there was light smoke in the control room. When operators shifted the ventilation system to the purge mode, the smoke increased to a point which required airline breathing equipment to be used. Investigation showed that the design of the control room habitability system, its technical specifications, and its surveillance and operating procedures did not consider an event such as this fire which produced heavy smoke in an area adjacent to the control room. The licensee has subsequently made modifications to correct the problem.

The AIT inspection was conducted October 3-6, 1987. The inspection findings are contained in AIT Inspection Report 50-267/87-26, which was forwarded to the licensee on October 29, 1987 (Ref. C-1). The AIT concluded that the plant remained within its design basis (i.e., both trains of the safe shutdown cooling remained operable - reference 10 CFR Part 50, Appendix R) throughout the fire and its aftermath. It was also concluded that the operators generally responded well and that the fire was quickly and effectively extinguished.

There was no release of radioactivity and the threat to public health or safety was minimal. The event received nationwide media coverage.

* * * * * * * *

2. Reactor Coolant Pump Shaft Cracking at Palo Verde Unit 1

During October 15-20, 1987, ultrasonic testing showed indications that the shafts of three reactor coolant pumps (RCPs) at Palo Verde Unit 1 were experiencing cracking in the area of the shaft keyways. The Unit has four RCPs. During November, more detailed inspections revealed cracks on all four RCP shafts. Palo Verde Units 1, 2, and 3 are pressurized water reactors, designed by Combustion Engineering (CE); there are four RCPs for each unit. The facility which is operated by Arizona Public Service Company, is located in Maricopa County, Arizona.

The RCP shafts were manufactured by a West German firm, KSB, and supplied to Palo Verde via CE. The licensee had been informed through CE that numerous European facilities, with RCPs similar in design to those at Palo Verde, had experienced shaft cracking. Two European RCPs had actually failed after 41,500 and 47,500 hours, respectively, of run time. By letter, dated October 8, 1987 (Ref. C-2), the licensee informed the NRC of their intention to inspect the Unit 1 shafts during the unit refueling outage which had commenced on October 2, 1987.

The shaft cracking is apparently due to the shaft material exceeding fatigue limits. The root cause of crack initiation has been attributed to a reduction in fatigue strength due to chrome plating in high stress areas of the shaft.

Once a crack has initiated, the crack apparently propagates slowly in a circumferential manner over millions of stress cycles.

At the time that the problem was identified in Unit 1, Unit 2 was operating at full power and Unit 3 was preparing for initial criticality. A meeting was held on October 24, 1987 between NRC and licensee personnel to discuss the information available and to consider the continued operation of Unit 2. Although the failure of one RCP is an analyzed accident, concerns were raised that the probability of a shaft failure had significantly increased and that there may be a potential for multiple shaft failures.

On October 25, 1987, a confirmatory order was issued by the NRC (Ref. C-3) which stated that continued operation of Unit 2 was acceptable, contingent upon the licensee providing increased monitoring of pump vibration. European experience indicates that increased shaft vibration occurs approximately two days prior to shaft failure. Therefore, close monitoring of pump vibration should provide sufficient time to shut down the unit prior to shaft failure.

The licensee is replacing the Palo Verde Unit 1 shafts with modified shafts during the unit's refueling outage. Modified shafts will also be installed in Units 2 and 3 during their first refueling outages.

* * * * * * * *

3. Numerous Complications During Recovery from a Loss of Offsite Power Event at Pilgrim

On November 12, 1987, with the Pilgrim reactor in cold shutdown, a severe winter snowstorm resulted in the loss of both offsite power sources to the station. Equipment problems led to the decision to manually trip one of Pilgrim's two diesel generators (DGs). This removed power to the one remaining source supplying instrument air. Although this did not present a safety concern, it was a distraction to the operators as air-operated equipment began to fail. Due to further complications, equipment failures and equipment being out of service, offsite power was not restored until about 21 hours after it was lost. Pilgrim, which is operated by the Boston Edison Company (the licensee), is a General Electric-designed boiling water reactor located in Plymouth County, Massachusetts.

At the time of the event, the reactor had been in cold shutdown for a period of about 19 months. Therefore, even though there was fuel in the reactor, the decay heat being generated was relatively low. This considerably reduced the significance of the event because even had all ac power been lost (together with the complications and delays encountered during the event), the risk of fuel damage was very low. Further details of the event are as follows.

Early on November 12, 1987, a severe winter storm was in progress consisting of gale force winds and heavy wet snow. At 2:05 a.m., fault tripping of the switchyard offsite power supply breakers occurred, isolating the station from the offsite ac transmission lines. The cause of the fault tripping has been attributed to snow and ice packing the high voltage insulators causing flashovers, and winds bringing the offsite power lines into proximity to one another. Both DGs "A" and "B" started and assumed loads from the emergency buses.

The fault tripping of the offsite power sources apparently caused the startup transformer differential protection to operate which indicated a possible internal fault in the transformer. This transformer is the preferred power source for outage conditions and was the only source available for quick recovery of offsite power. Because the reactor decay heat was so low, the licensee believed there was no necessity to restore offsite power quickly and jeopardize plant equipment or personnel. To insure the transformer was safe to re-energize, the transformer was tested for internal faults by meggering, high potential testing, and oil sample analysis. All these tests, which took several hours, ultimately showed the transformer to be undamaged. In addition, because the high voltage insulators had become packed with snow, the high voltage switchyard required a wash-down before reenergizing.

Further complications developed during the period the startup transformer was out of service for testing. The "B" DG output current transformer supplying generator control and protective equipment indicated an open circuit. This presented a potential hazard to the operation of the diesel and the decision was made to secure the "B" DG. This left the station with only one source of ac power (i.e., DG "A"). As previously mentioned, shutting down the "B" DG removed power to the one remaining source supplying instrument air. (Two of the three emergency bus supplied instrument air compressors were out of service at the time for maintenance.) This caused air operated equipment to begin to fail. While this was not a safety concern, it was a distraction to the operators, and added to the complications encountered during recovery operations.

Since the startup transformer was expected to be out of service for some time for the previously mentioned testing, the licensee explored alternate means of restoring offsite power. One alternate, the 23 kV shutdown transformer, was out of service due to modifications in progress to install a third DG at the station. The licensee decided that restoring this transformer was not feasible due to the extent of work which would be necessary.

The other alternate method was through backfeeding of the main and the unit auxiliary transformers, which required some time to remove bus links between the main generator and the transformers, and making other changes. Ultimately, this was the method used which first restored offsite power to the site at 11:09 p.m. on November 12, 1987, about 21 hours after power was lost. However, if power had to be restored quickly, it was estimated that the earliest recovery time would have been 11 to 12 hours after power loss.

DG "B" was not declared fully operational until 11:15 p.m. on November 14, 1987. This delay was due primarily to equipment failures including binding of the prelubrication pump shaft (which had to be replaced) and leaking fuel injectors (which had to be repaired). The failures appear to have resulted from inadequate maintenance performed prior to this loss of offsite power event.

An additional complication encountered during the recovery efforts was a delay in reestablishing shutdown cooling. This was due to blown fuses in the analog trip system. The root cause for the blown fuses remains under investigation.

The NRC sent an Augmented Inspection Team (AIT) to perform an inspection at the site during the period of November 16-20, 1987. This was considered necessary because (a) the Pilgrim station has experienced numerous loss of offsite power events, (b) the numerous complications encountered during the recovery efforts from the November 12, 1987 event, and (c) the extended period of time the station was without offsite power. The conclusions and recommendations of the AIT will be issued as NRC Inspection Report No. 50-293/87-53.

The licensee continues to develop necessary corrective actions. Additional actions are also expected to be made as a result of the AIT inspection. Meanwhile, on November 18, 1987, the licensee had committed to several plant improvements prior to reactor startup: (a) completion of the installation of the third DG, which is already on site, to enhance electrical supply reliability; (b) installation of an additional backup instrument air supply; and (c) installation of additional instrumentation to monitor switchyard conditions to allow rapid evaluation of any abnormal conditions and to assist in the restoration of power.

* * * * * * * *

REFERENCES
FOR APPENDICES

B-1 U.S. Nuclear Regulatory Commission, "Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2, Docket No. 50-320," USNRC Report NUREG-0683, Supplement No. 2 (Final Supplement Dealing with Disposal of Accident-Generated Water), published June 1987.**

B-2 Letter from James M. Taylor, Director, NRC Office of Inspection and Enforcement, to William Cavanaugh, III, President, Mississippi Power and Light Company, forwarding a Notice of Violation and Proposed Imposition of Civil Penalties, Docket No. 50-416, June 3, 1985.*

B-3 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to Babcock & Wilcox Owners Group Plant Reassessment Program," NUREG-1231, issued November 1987.**

C-1 Letter from L. J. Callan, Director, Division of Reactor Projects, NRC Region IV, to Robert O. Williams, Jr., Vice President, Nuclear Operations, Public Service Company of Colorado, forwarding AIT Inspection Report 50-267/87-26, Docket No. 50-267, October 29, 1987.*

C-2 Letter (Corrected Copy) from J. G. Haynes, Vice President, Nuclear Production, Arizona Nuclear Power Project, to U.S. Nuclear Regulatory Commission, Document Control Desk, Docket Nos. 50-528, 50-529 and 50-530, October 8, 1987.*

C-3 Letter from George W. Knighton, Director, Project Directorate V, Division of Reactor Projects - III, IV, V, and Special Projects, NRC Office of Nuclear Reactor Regulation, to E. E. Van Brunt, Jr., Executive Vice President, Arizona Nuclear Power Project, forwarding "Order Modifying License Confirming Licensee Commitments on Monitoring Vibration of Reactor Coolant Pump Shafts at Palo Verde Nuclear Generating Station, Unit No. 2, Effective Immediately," Docket No. 50-529.*

---

*Available in NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555, for public inspection and/or copying.

**Available for purchase from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082. Washington, DC 20013-7082. Also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for public inspection and/or copying at the NRC Public Document Room, 1717 H Street, NW, Washington, DC.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any): NUREG-0090 Vol. 10, No. 4

2. TITLE AND SUBTITLE: Report to Congress on Abnormal Occurrences October - December 1987

3. LEAVE BLANK

4. DATE REPORT COMPLETED: MONTH March, YEAR 1988

5. AUTHOR(S)

6. DATE REPORT ISSUED: MONTH March, YEAR 1988

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code):
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code): Same as 7., above

11a. TYPE OF REPORT: Quarterly

b. PERIOD COVERED (Inclusive dates): October - December 1987

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 to December 31, 1987. During the report period, there was one abnormal occurrence at the NRC licensees; the item involved the suspension of license of an oil and gas well tracer company for noncompliance with NRC regulatory requirements. There were no abnormal occurrences reported by the Agreement States. The report also contains information updating some previously reported abnormal occurrences.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS: Suspension of License; Oil and Gas Well Tracer Company; Turbine Building Fire; Reactor Coolant Pump Shaft Cracking; Loss of Offsite Power.

b. IDENTIFIERS/OPEN ENDED TERMS

15. AVAILABILITY STATEMENT: Unlimited

16. SECURITY CLASSIFICATION: (This page) Unclassified; (This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
PERMIT No. G-67

NUREG-0090, Vol. 10, No. 4 REPORT TO CONGRESS ON ABNORMAL OCCURRENCES MARCH 1988