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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 October 1 to December 31, 1982.

The report states that for this report period, there was one abnormal occurrence at the NRC licensees. The event involved the containment spray system being inoperable at one of the nuclear power plants licensed to operate. The Agreement States reported no abnormal occurrences to the NRC.

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

CONTENTS

	Page
ABSTRACT	iii
PREFACE	vii
INTRODUCTION	vii
THE REGULATORY SYSTEM	vii
REPORTABLE OCCURRENCES	viii
AGREEMENT STATES	ix
FOREIGN INFORMATION	x
REPORT TO CONGRESS ON ABNORMAL OCCURRENCES, OCTOBER-DECEMBER 1982.	1
NUCLEAR POWER PLANTS	1
82-7 Inoperable Containment Spray System	1
FUEL CYCLE FACILITIES (Other than Nuclear Power Plants)	4
OTHER NRC LICENSEES (Industrial Radiographers, Medical Institutions, Industrial Users, Etc.)	4
AGREEMENT STATE LICENSEES	4
REFERENCES	5
APPENDIX A - ABNORMAL OCCURRENCE CRITERIA	7
APPENDIX B - UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES ..	11
NUCLEAR POWER PLANTS	11
75-5 Cracks in Pipes at Boiling Water Reactors (BWRs)	11
79-3 Nuclear Accident at Three Mile Island	12
OTHER NRC LICENSEES	18
82-6 Radiological Contamination from Well Logging Operations	18
APPENDIX C - OTHER EVENTS OF INTEREST	19
REFERENCES (FOR APPENDICES)	27

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 between October 1 to December 31, 1982.

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. Requirements for reporting incidents or events exist which help identify deficiencies early and aid in assuring that corrective action is taken to prevent their recurrence.

After the accident at Three Mile Island in March 1979, the NRC and other groups (a Presidential Commission, Congressional and NRC special inquiries, industry, special interests, etc.) spent substantial efforts to analyze the accident and its implications for the safety of operating reactors and to identify the changes needed to improve safety. Some deficiencies in design, operation and regulation were identified that required actions to upgrade the safety of nuclear power plants. These included modifying plant hardware, improving emergency preparedness, and increasing considerably the emphasis on human factors such as expanding the number, training, and qualifications of the reactor operating staff and upgrading plant management and technical support staffs' capabilities. In addition, each plant has installed dedicated telephone lines to the NRC for rapid communication in the event of any incident. Dedicated groups have been formed both by the NRC and by the 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 the licensing and regulation process.

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

Since the NRC is responsible for assuring that regulated nuclear activities are conducted safely, the nuclear industry is required to report incidents or events which involve a variance from the regulations, such as personnel overexposures, radioactive material releases above prescribed limits, and malfuntions of safety-related equipment. Thus, a reportable occurrence is any incident or event occurring at a licensed facility or related to licensed activities which NRC licensees are required to report to the NRC. The NRC evaluates each reportable occurrence to determine the safety implications involved.

Because of the broad scope of regulation and the conservative attitude toward safety, there are a large number of events reported to the NRC. The information provided in these reports is used by the NRC and the industry in their continuing evaluation and improvement of nuclear safety. Some of the reports describe events that have real or potential safety implications; however, most of the reports received from licensed nuclear power facilities describe events that did not directly involve the nuclear reactor itself, but involved equipment and components which are peripheral aspects of the nuclear steam supply system, and are minor in nature with respect to impact on public health and safety. Many are discovered during routine inspection and surveillance testing and are corrected upon discovery. Typically, they concern single malfunctions of components or parts of systems, with redundant operable components or systems continuing to be available to perform the design function.

Information concerning reportable occurrences at facilities licensed or otherwise regulated by the NRC is routinely disseminated by NRC to the nuclear industry, the public, and other interested groups as these events occur. Dissemination includes deposit of incident reports in the NRC's public document rooms, special notifications to licensees and other affected or interested groups, and public announcements. In addition, a computer printout containing information on reportable events received from NRC licensees is routinely sent to the NRC's more than 100 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

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

AGREEMENT STATES

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

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

In early 1977, the Commission determined that abnormal occurrences happening at facilities of Agreement State licensees should be included in the quarterly report to Congress. The abnormal occurrence criteria included in Appendix A 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 maded in these quarterly abnormal occurrence reports to Congress; however, only domestic abnormal occurrences are reported.

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

OCTOBER-DECEMBER 1982

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the fourth calendar quarter of 1982. As of the date of this report, the NRC had determined that the following was an abnormal occurrence.

82-7 Inoperable Containment Spray System

The following information pertaining to this event is also being reported in the Federal Register (Ref. 1). Appendix A (see general criterion 2) of this report notes that major degradation of essential safety-related equipment can be considered an abnormal occurrence. In addition, Example 3 under "For Commercial Nuclear Power Plants" of Appendix A notes that 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 can be considered an abnormal occurrence.

Date and Place - On October 28, 1982, Alabama Power Company (the licensee) notified the NRC that the manual isolation valves for both train A and train B of the containment spray system were found locked in the closed position at their Farley Nuclear Plant Unit 2. Farley Unit 2 utilizes a Westinghouse designed pressurized water reactor and is located in Houston County, Alabama.

Nature and Probable Consequences - Farley Unit 2 was taken to cold shutdown on October 24, 1982 to begin a refueling and maintenance outage. On October 28, 1982, while aligning valves for certain scheduled inservice inspections, the licensee found the containment spray header isolation valve on each of the two supply headers locked in the closed position. These valves, located inside the Unit 2 containment building, supply separate, redundant, containment spray rings. After investigation and record searches of valve movement documentation, the licensee concluded that the valves had been closed since before the plant achieved initial criticality on May 8, 1981. Thus the redundant, containment spray systems were inoperable during this period and consequently would have been unable to fulfill their safety function.

The safety function of the containment spray system is to spray borated water into the containment to limit the maximum pressure in the containment to less than the design pressure following certain steam line breaks or loss of coolant accidents and to reduce the pressure and temperature to minimize containment leakage. The sytem is also designed to spray sodium hydroxide into the containment to remove radioactive iodine which would limit iodine doses to less than 10 CFR Part 100 limits should a LOCA occur.

The plant also has a containment fan cooler system, which is used during normal operation to recirculate and cool the containment atmosphere. Following a LOCA or steam line break accident, the system acts in conjunction with the containment spray system to reduce containment temperature and pressure. The amount of pressure and temperature reduction depends upon the number of containment spray rings and fan coolers that would operate following such an accident. The licensee's technical specifications require a minimum of one containment spray system and one fan cooler to be operable. As discussed below, the containment fan cooler system working alone, even with only one fan operable, can be expected to protect the integrity of the containment and the safety equipment inside. However, the containment fan cooler system does not have the radioactive iodine removal capabilities of the containment spray system.

Conservative calculations were made by the NRC and the licensee to determine the effect on containment pressure, containment temperature, and iodine doses had a LOCA or a main steam line break (MSLB accident occurred while the containment sprays were inoperable.

In regard to containment pressure, the most limiting accident would be a MSLB of 0.7 square feet at 30% power with a single failure of the containment fan coolers. With two out of four fan coolers in operation, the calculated peak pressure would be 55.1 psig. With only one fan cooler in operation (based on the plant's technical specifications requiring only one fan cooler per train such that the worst single failure would result in only one fan cooler being operational), the analysis predicts a peak containment pressure of 61.6 psig. Both calculated pressures are higher than the containment design pressure of 54 psig. However, even for the more conservative calculation, containment integrity would likely be maintained since the containment has been tested at 62.1 psig.

Peak containment temperature, based on the most limiting MSLB, was conservatively calculated by the licensee to compare to the equipment qualification temperatures. Generally, the calculated peak temperature exceeded the qualification temperatures by less than 20°F. In one case, the difference was about 50°F. However, the required operating times for many components are short and the thermal lag inside the equipment housings would be expected to preclude damage to the internal components prior to performing their specified functions.

The radiological consequences at both the exclusion area and the low population zone boundaries were conservatively calculated based on a LOCA and rupture of fuel cladding. Calculations were made by the NRC staff for the maximum allowable containment leak rates permitted by the licensee's technical specifications and for the leak rate as measured at the plant when last tested. In both cases, analyses indicate that thyroid doses would exceed 10 CFR Part 100 limits at both the exclusion area and the low population zone boundaries.

The licensee also made calculations based on what the licensee considered more "realistic" assumptions. The licensee concluded that offsite exposures could be expected to be less than 10 CFR Part 100 guideline values, based on the "realistic" assumptions. However, since the valves had been closed since before initial plant startup, variations could be expected in such parameters as containment leak rates (last performed and reported to the NRC in mid-1980) and meteorological conditions.

The importance of the event is emphasized since the subject valves at Farley are located inside containment and are manually operated. Therefore, in the event of a LOCA or a MSLB accident, the valves would not be accessible to be opened by plant personnel.

Cause or Causes - The event was caused by the valves not being in conformance with design drawings and by a procedural inadequacy used for operator determination of valve position. A unique condition developed in these valves when the vendor, Westinghouse, made a design change that lengthened the valve stem to increase the valve's adaptability to a motor operated valve (however, as described above, the valves are manually operated at Farley). The design change resulted in a valve stem that makes the valve appear to be open when it is actually closed. That is, in the closed position, the extra long valve stem shows six inches of threaded stem extending out of the bonnet.

Therefore, operators, who were instructed and trained to observe valve stem positions in order to verify the valve positions, erroneously interpreted these valves as being open when they were, in fact, closed. However, similar valves were found in the correct (locked open) position in Unit 1. This indicates that operator error may have been a contributing factor to this event.

In addition, Westinghouse did not provide revised drawings showing the valve modification. As a result, the overall dimension of the installed valve stem was six inches longer that that specified.

Actions Taken to Prevent Recurrence

Licensee - Alabama Power has obtained concurrence from Westinghouse Corporation to cut the excess stem off the valves so as to conform with design drawings and with other rising stem gate valves throughout the plant. In addition, as a further safeguard to prevent recurrence, plant administrative procedures covering valve position verification have been changed to require that manual valves which are locked open will be moved in the shut direction to verify their position; then the valve will be returned, if applicable, to the original position.

As stated previously, the licensee performed analyses of the effects on containment pressure, containment temperature, and iodine dosages had a design basis accident occurred while the containment sprays were inoperable. These analyses were submitted to the NRC for review on November 30, 1982 and December 3, 1982 (Refs. 2 and 3).

After the locked valves were found on Unit 2, the licensee checked the containment spray valves on Unit 1. The valves were found to be locked open as required. Since the Unit 1 valves are identical to those of Unit 2, the corrective actions described above are applicable to both units.

NRC - As stated previously, the NRC performed conservative analyses of containment pressure and iodine release to compare to the licensee's analyses.

An enforcement conference was held in the NRC Region II (Atlanta) office with the licensee on November 19, 1982 (Ref. 4). The licensee presented their program for preventing recurrence. The NRC concurred with the licensee's corrective actions.

NRC Region II performed inspections to determine the circumstances associated with this event. Based on these inspections, a Notice of Violation and Proposed Imposition of Civil Penalty (for $40,000) was issued to the licensee on February 2, 1983 (Ref. 5). The licensee paid the civil penalty on February 28, 1983.

This incident is closed for purposes of this report.

FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the fourth calendar quarter of 1982. 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 radioisotopes in the medical, industrail, 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 1982. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the fourth calendar quarter of 1982, the Agreement States reported no abnormal occurrences to the NRC.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Inoperable Containment Spray Systems," Federal Register. (Item is being published in the Federal Register concurrently with this report.)

2. Letter from F. L. Clayton, Jr., Senior Vice President, Alabama Power Company, to J. P. O'Reilly, Regional Administrator, NRC Region II, Docket No. 50-364, November 30, 1982.*

3. Letter from F. L. Clayton, Jr., Senior Vice President, Alabama Power Company, to J. R. O'Reilly, Regional Administrator, NRC Region II, Docket No. 50-364, December 3, 1982.*

4. Letter from R. C. Lewis, Director, Division of Project and Resident Programs, NRC Region II, to R. P. McDonald, Vice President-Nuclear Generation, Alabama Power Company, Docket No. 50-364, December 7, 1982.*

5. Letter from J. P. O'Reilly, Regional Administrator, NRC Region II, to R. P. McDonald, Vice President-Nuclear Generation, Alabama Power Company, forwarding a Notice of Violation and Proposed Imposition of Civil Penalty, Docket No. 50-364, February 2, 1983.*

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*Available in NRC Public Document Room, 1717 H Street, NW., Washington, D.C. 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 set forth in an NRC policy statement published in the FEDERAL REGISTER on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952).

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

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

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

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

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

<u>For All Licensees</u>

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 § 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 regulatory limit (10 CFR § 71.36(a)).

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 signficant 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 § 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 § 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 specifially 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 1982 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. Those occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

75-5 Cracks in Pipes at Boiling Water Reactors (BWRs)

This abnormal occurrence was originally reported in NUREG-75/090, "Report to the Congress on Abnormal Occurrences: January-June 1975," and updated (and previously closed out) in subsequent reports in this series, i.e, NUREG-0090-1; 0090-2; 0090-3; Vol. 1, No. 3; Vol. 2, No. 2; Vol. 2, No. 4; Vol. 3, No. 2; Vol. 3, No. 4; and Vol. 5, No. 2. It is being reopened to report the following new significant information.

NUREG-0900, Vol. 5, No. 2 included an update to describe cracks detected in the recirculation system piping at Nine Mile Point Unit 1. As reported, the licensee is replacing the 28-inch recirculation piping in all five recirculation loops and all ten safe ends; the replacement material is of a type less susceptible to intergranular stress corrosion cracking.

NRC Inspection and Enforcement Bulletin No. 82-03 (Ref. B-1) was issued on October 14, 1982, and Revision 1 on October 28, 1982 (Ref. B-2), requiring all BWRs that were shut down or scheduled to be shut down by January 31, 1983 to augment the normal inspections of the recirculation system piping; the licensees were also required to demonstrate the capability of their personnel and procedures for detecting very small cracks in pipe samples taken from Nine Mile Point Unit 1.

The licensee for Monticello examined all the welds in the recirculation system and connecting piping and, as a result, found indications of cracks in five welds. The flaws were repaired and the plant has resumed power generation. Subsequently, indications of cracks were found in seven welds at Hatch Unit 1 and indications were found in two welds in the large diameter piping in the recirculation system piping at Browns Ferry Unit 2. Crack indications were also recently found in weld locations in the reactor coolant recirculation system at Dresden Unit 2 (one weld) and Brunswick Unit 1 (three welds). The NRC is closely monitoring the licensees' corrective actions and making evaluations to assure that the plants will be safe to restart.

Further reports will be made as appropriate.

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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 subsequent reports in this series, i.e., NUREG-0090, Vol. 2, No. 2; Vol. 2, No. 3; Vol. 2, No. 4; Vol. 3, No. 1; Vol. 3, No. 2; Vol. 3, No. 3; Vol. 3, No. 4; Vol. 4, No. 1; Vol. 4, No. 2; Vol. 4, No. 3; Vol 4, No. 4; Vol. 5, No. 1; Vol. 5, No. 2; and Vol. 5, No. 3. It is further updated as follows.

Reactor Building Entries

It should be noted that the reactor building entries between August 30, 1982 and September 17, 1982 discussed below had been previously mentioned in the last update report (i.e., NUREG-0090, Vol. 5, No. 3). The information discussed below expands that presented in the previous report.

During the reactor building entries on August 30, 1982, September 1, 1982 and September 3, 1982, activities conducted included continued polar crane damage assessment, remote decontamination of the 282 ft. elevation, primary coolant sampling, housekeeping, and the installation of a manometer on the reactor vessel head to sample and measure the rate of gas generation in the reactor vessel. A closed circuit television inspection of the reactor building below the 305 ft. elevation was also made.

During the reactor building entries on September 8, 1982 and September 10, 1982, the most labor intensive tasks conducted involved polar crane damage assessment and continued remote decontamination of the 282 ft. elevation. The weekly primary system water sample was taken, and a gas sample was collected from the center control rod drive mechanism (CRDM), to determine the composition and the generation rate of gases from the core. The center CRDM had been isolated and inerted with nitrogen on September 3, 1982. A "Base Line" gas sample that was taken then indicated hydrogen below detectable limits, nitrogen at 95.5% and oxygen at 4.3%. Gas sample measurements on September 8, 1982 indicated a gas generation rate of 0.06 cubic feet per day; the gas sample indicated hydrogen at 6.3%, nitrogen at 87.4%, oxygen at 4.3% and other gases at 2%.

During the entries on September 15 and 17, 1982, portions of the reactor building dome were sprayed with a water jet, heated to 140°F, to remove loose surface contamination. Additional entry tasks included continued remote decontamination of the 282 ft. elevation and general housekeeping. A primary system gas sample was taken from the center control rod drive mechanism, indicating that the gas generated in the core was not collecting in explosive concentrations. The sample indicated that hydrogen gas was being released, but there did not appear to be any release of oxygen to support combustion. Based on these measurements, the gas generation rate in the reactor vessel was calculated to be less than 0.02 cubic foot per day.

During the entries on September 20, 22, and 24, 1982, decontamination of the reactor building dome continued. The dome decontamination is the first phase of an ongoing decontamination effort to reduce loose surface contamination on exposed reactor building surfaces. As previously discussed, high pressure, hot water spray was used as the basic decontamination technique in the reactor building during subsequent phases of the decontamination. It is anticipated that one to two hundred thousand gallons of previously processed water will be used. The water is collected in the reactor building sump for reprocessing through the Submerged Demineralizer System (SDS).

The decontamination was performed over a period of several months. The reactor building purge was operated almost continuously during the decontamination to maintain building temperatures at about 60°F to minimize heat stress on personnel working inside. Continuous operation of the purge is not expected to significantly increase offsite releases. Radioactive particulate material in the purge air flow was effectively removed by passing through three sequential sets of filters. The first filter, called a roughing filter, is similar to a home furnace filter; it keeps the two downstream filters physically clean. The second and third filters are called HEPA (high efficiency particulate air) filters; they remove essentially all of the remaining particulate material in the air flow. (Two HEPA filters are used to provide system redundancy.) To date, there has been no indication of increased particulate releases during reactor building entry purges.

Decontamination of the reactor building dome and polar crane continued during the entries on September 27, 29, and October 7, 1982. Gas samples were taken from the pressurizer and reactor coolant system high points to determine whether hydrogen gas was accumulating in the primary system. The analysis indicated that hydrogen gas concentrations were below the combustible limits. A brief inspection of components and wall surfaces below the 305 ft. elevation was conducted using a closed circuit television camera. A "dirt ring," showing the basement high-water level, was visible on the D-ring wall.

Four reactor building entries were conducted during the week of October 10, 1982, primarily for further decontamination activities. The spraying of reactor building interior surfaces added approximately 60,000 gallons of processed water to the reactor building sump. Thirty thousand gallons of this water were subsequently transferred to the SDS feed tanks for reprocessing. Spraying of the reactor building dome has been completed. The current decontamination effort is focused on remote spraying below the 305 ft. elevation and a manual decontamination of the polar crane.

In conjunction with these decontamination activities, a closed circuit television inspection of the 282 ft. elevation was made. A prominent "dirt ring" approximately one foot wide was visible on vertical surfaces. The elevation of the "dirt ring" appears to correspond to the elevation of the reactor building high water level (291 ft.). An examination of the 282 ft. floor surfaces was masked by approximately four inches of water which accumulated in the reactor building basement from decontamination activities. The water appeared relatively clear, but its depth distorted the view of the floor surface. An inspection of the reactor coolant drain tank cubicle, which included an inspection of the rupture disk discharge pipe, did not identify any component damage.

Four reactor building entries were conducted during the week of October 17, 1982. Decontamination of the polar crane was the predominant in-containment activity. The polar crane decontamination techniques include the use of a mild chemical degreaser, hands-on decontamination, vacuuming, and flushing. Following the decontamination, a strippable coating was applied to the crane surfaces to help control surface contamination.

During the week of October 24, 1982, four reactor building entries were conducted, and the hands-on decontamination of the polar crane was completed.

Polar crane refurbishment was the predominant in-containment activity for the three reactor building entries that were conducted during the week of October 31, 1982. The main electrical power line was attached to the polar crane and functional checks of control circuits were started. Decontamination of the reactor building is continuing in parallel with the polar crane refurbishment.

During the week of November 7, 1982, three entries were conducted in support of polar crane refurbishment activities.

Reactor building entries were conducted on November 15, 17, 18, and 19, 1982. In addition to continuing the polar crane refurbishment (which has been identified as the critical path and priority activity), the following tasks were performed in the reactor building: three leadscrews, which could not be uncoupled from their control rods on the first attempt, were uncoupled; the procedure to raise the axial power shaping rod (APSR) leadscrews, in preparation for head lift, was initiated; and leadscrew 8H, which had been removed from the reactor during the quick look inspection, was cut and segments removed from the reactor building for eventual shipment offsite for analysis. Reactor building decontamination using high pressure, high temperature water, limited "hands-on" decontamination, and some strippable coating application continued in parallel with the other activities in the reactor building.

Four reactor building entries were conducted in the week following the Thanksgiving holiday. In the polar crane refurbishment program, the four slow speed bridge drive motors were mechanically uncoupled from the load and electrically activitated. The motors operated normally and appeared to be satisfactory for driving the polar crane bridge.

A total of eighteen reactor building entries were made during the month of December 1982. Polar crane refurbishment and reactor building decontamination were the most man-hour intensive tasks in the reactor building during this period.

During the week of December 5, 1982, in addition to the on-going decontamination and polar crane refurbishment, considerable effort was expended in preparing the control rod drive system for eventual reactor vessel head removal. All eight axial-power-shaping-rod leadscrews were raised to their parked positions. (The parked position is in the upper portion of the control rod drive assembly; it ensures that the leadscrew is above the reactor vessel flange and clear of potential interference during reactor vessel head removal.)

On December 17, 1982, the reactor (the primary side of the system) was refilled and pressurized to 70 psig. Primary system refill is a prerequisite for refilling the steam generators (the secondary side of the system) in preparation for chemical conditioning of the steam generator secondary water.

Prior to refilling, temperature and radiation probes were lowered into the reactor vessel. The temperature in the core region, above the rubble bed and in the lower two feet of the plenum, was 107°F. The temperature in the upper portions of the plenum was 102°F. The gamma-sensitive radiation probe was lowered into the vessel to a height of approximately six inches below the top of the plenum. This elevation corresponds to the approximate elevation of the reactor vessel head flange. The probe was lowered through control rod drive leadscrew openings at the core periphery and at a location midway from the periphery and the core center. At both locations, probe measurements indicated the radiation levels near the upper surface of the plenum ranged from 520 to 600 R/hr. The measurements were made under water inside the 8 1/2-inch diameter control rod guide tubes. Radiation levels four feet above the plenum were 50 R/hr at the core periphery and 120 R/hr midway between the core periphery and center. The radiation data are being evaluated to determine possible impact on reactor vessel head removal.

The secondary side of the "A" steam generator was filled on December 20, 1982, as the first step in the chemical conditioning process which will establish recommended long term lay-up chemistry in the steam generators. The process involves recirculation of chemically treated water through the secondary system prior to draining the steam generators for long term lay-up.

EPICOR II Prefilter Shipments

On October 7, 1982, the third in a group of 49 EPICOR II prefilters (PF-2) was shipped from TMI to the Idaho National Engineering Laboratory (INEL) in Scoville, Idaho. The PF-2 liner and shipping cask (CNS-120-3) were inerted with nitrogen as an added safety precaution to ensure that no combustible gases will exist during shipment. The hydrogen-oxygen composition in the liner will be maintained at less than 2.5% hydrogen and less than 0.5% oxygen.

Two EPICOR II prefilter shipments were made from TMI to INEL on October 20 and 23, 1982, respectively: PF-7 in the CNS-8-120 cask and PF-8 in the HN-200 cask. One EPICOR II prefilter shipment (PF-9) was made from TMI to INEL on October 28, 1982. Two EPICOR II prefilter shipments were made from TMI to INEL on November 2 and 3, respectively: PF-45 in the CNS-9-120 cask and PF-46 in the HN-200 cask; in both shipments, the EPICOR liner and shipping cask cavity were inerted with nitrogen gas.

EPICOR II prefilter liner PF-20 was shipped from TMI to INEL on November 17, 1982 in a CNS-8-120 type B shipping cask; this liner was the ninth in a group of 49 EPICOR liners to be shipped to INEL. EPICOR II prefilter liners PF-47 and PF-27 were shipped from TMI to INEL on November 29 and December 1, 1982, respectively. EPICOR prefilter PF-48 was shipped to INEL on December 6, 1982. PF-6, 18, and 44 were shipped during the week of December 12, 1982. With the addition of the shipment of PF-49 on December 29, 1982, a total of 16 in a group of 49 EPICOR prefilters were shipped to INEL during 1982.

EPICOR II/SDS Processing

The EPICOR II system began processing SDS effluent (SDS Batch No. 36, EPICOR II Batch 143) on September 30, 1982. Approximately 5,000 gallons were processed. On October 3, 1982, processing of SDS Batch 37 began; this was completed on October 7, 1982 with approximately 5,000 gallons being processed.

SDS began processing of Batch No. 38 (approximately 44,000 gallons of reactor building sump water) on November 6, 1982. This water, which was previously processed by the SDS, was reused for the ongoing decontamination activities in the reactor building, and collected in the reactor building sump.

The EPICOR II system was activated on November 10, 1982 to process SDS effluent from Batch No. 38.

The reactor coolant system (RCS) feed and bleed process was resumed and completed December 13, 1982 after which the RCS was refilled and pressurized. SDS processing of Batch 39 (approximately 40,000 gallons) began December 18. After approximately eight hours of operation, a radiation level monitor alarmed, and the system was shut down. High levels of radiation were not found. However, one system component was replaced, and maintenance was necessary on another component before the system was restarted.

Processing of Batch 6 of RCS water (SDS Batches 39 and 40), which had been temporarily halted on December 18 for component repair and maintenance, was resumed on December 18 and completed on December 27, 1982. Since then, another "feed-and-bleed" process has been performed and 40,000 gallons of water from the reactor building sump were staged in preparation for additional SDS processing.

Groundwater Sampling Program

Periodic sampling of TMI groundwater began in January 1980 in an effort to detect any potential leakage from the contaminated water in the basement of the reactor building. When the SDS began processing reactor building water, the basement contained approximately 600,000 gallons of highly radioactive water (greater than 150 μCi/ml). There was a concern, if the reactor building leaked, that the leakage could contaminate Three Mile Island groundwater. However, the monitoring program has accumulated data to indicate that there was no leakage from the reactor building. The program did identify some groundwater contamination which resulted from leakage from the borated water storage tank (BWST).

The possibility of groundwater contamination from the potential sources of leakage has been reduced. Except for a periodic addition of water from ongoing reactor building decontamination, the water in the reactor building has been removed. A leakage collection trough and more sensitive level indicating equipment have been added to the BWST. The effectiveness of these measures will continue to be evaluated by the groundwater monitoring program.

Pre-TMI monitoring data indicate that surface water, drinking water, and precipitation in the TMI area will contain an average of 300 pCi/L of tritium with values as high as 600 pCi/L within the expected range. The highest TMI groundwater contamination was recorded in test boring 17 on March 23, 1982 (1.1×10^6 pCi/L). Test boring 17 is an area considered as restricted; the maximum permissible concentration for tritium in restricted areas is 0.1 μCi/ml (1×10^8 pCi/L).

Tritium was the predominant radioisotope detected in the groundwater. However, sporadic trace levels of radioactive cesium have been detected in test boring 2. On June 1, 1982, 11 pCi/L of antimony-125 was detected in test boring 17. (This antimony concentration was just above the lower limit of detection.) Subsequent samples from test boring 17 did not contain any detectable antimony.

Advisory Panel for the Decontamination of TMI-2

On November 17, 1982, the Advisory Panel for the Decontamination of TMI-2 held a public meeting at the Holiday Inn in Harrisburg, Pennsylvania. The panel received an update of the cleanup progress from GPU as well as status reports from the NRC, EPA, and DOE. The panel viewed the video tapes of the Unit 2 reactor core inspections that were performed in July and August 1982. Additional topics of discussion were the funding situation, cleanup schedules, accident generated water disposition, and transportation routing of radioactive waste shipments. The next Panel meeting will be on February 2, 1983 in Harrisburg, Pennsylvania.

Fire Hazards Evaluation

As part of the NRC evaluation of the plant fire hazards and fire protection, a reactor building entry, which included an NRC employee and a contractor, was made on December 2, 1982. The entry team traversed all levels of the building, except the highly contaminated basement, observing the condition of fire hoses and fire extinguishers and looking for possible fire hazards created by material and equipment that had been brought into the building. No significant hazards or fire protection equipment deficiencies were identified.

Further reports will be made as appropriate.

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OTHER NRC LICENSEES

82-6 Radiological Contamination from Well Logging Operations

This abnormal occurrence was originally reported in NUREG-0090, Vol. 5, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1982." It is updated as follows.

The licensee hired a radiological safety consultant to supervise the cleanup of the contaminated land and equipment and the packaging and disposal of all radioactive waste. Overall decontamination operations are approximately 80% complete, as of January 1, 1983. Cleanup of the small contaminated area (identified as a second site nearby in the original report) near Pine Bank, Pennsylvania is complete and the licensee has requested permission to release this area for unrestricted use. NRC Region I is currently evaluating this request. The licensee has generated over 400 55-gallon drums of contaminated soil and other material. The original hole where the source was ruptured has been cleaned and filled with concrete. The licensee projects that decontamination will be complete early in 1983. NRC Region I is closely monitoring the cleanup and will conduct a comprehensive closeout survey prior to release of the site for unrestricted use.

Further reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

The following events are described below because they may be perceived by the public to be of public health and safety significance. They were determined not reportable as abnormal occurrences.

1. <u>Control Rod Drive Failure and Reactor Trip</u>

On September 30, 1982, Commonwealth Edison Company (the licensee) experienced a control rod insertion problem at their Zion Unit 1 plant while the plant was operating at full power. The control rods would not move into the reactor core in the normal operational mode, position-by-position. The capability to and automatically. Zion Unit 1 utilizes a Westinghouse designed pressurized water reactor and is located in Lake County, Illinois.

At about 4:45 p.m., one of two main feedwater pumps failed because of a limited non-safety related power failure in the auxiliary building. The operators immediately ran the turbine back to 50% power in an effort to keep the reactor from tripping. The control rod drive system, which should have automatically stepped the control rods inward in response to the increasing reactor coolant temperature, failed to do so. The operator then attempted to insert rods in the manual mode; however, the rods still did not move. Seeing that the primary system pressure and temperature were still increasing, and that the control rods were not responding, the shift engineer ordered a manual trip of the reactor. This successfully occurred at 4:50 p.m.

The power failure in the auxiliary building also had disabled the steam (turbine) bypass valves, which would normally divert steam directly to the condenser. Thus, with these valves inoperable, and the turbine valves closed by the reactor trip, the heat in the primary system and the increasing pressure in the secondary system could only be released via the steam generator code safety valves. Accordingly, all 20 safety valves lifted for approximately 30 seconds.

Immediately after the reactor trip, operators observed that there was no bottom light indication for five of the control rods. In accordance with approved opearting procedures, the operators commenced emergency boration of the reactor coolant system until the faulty rod bottom lights and position indicators were corrected and all rods were verified to be inserted. The emergency boration lasted about 6 minutes. Within 3 minutes after the reactor trip, power to the steam dump valves was restored, making them available for decay heat removal. The plant was maintained in hot shutdown pending evaluation of the various problems identified.

The peak reactor coolant system temperature, pressure, and pressurizer level recorded during the transient were 581°F, 2355 psig, and 62%, respectively; no plant safety limits were exceeded. The downstream temperature sensors indicated that at least one primary system power-operated relief valve (PORV) had lifted during the event. Since there was no discernable increase in the primary relief tank temperature, it was concluded tht the opening of the PORV was of very short duration. The day after the transient, some iodine-131 activity was detected in a steam generator sample. This indicated that the transient may have opened a small primary to secondary leak.

The rod control system failure was found to be due to a malfunction in its circuitry. This control feature is not safety related. However, the control rod scram circuits, which are safety related, remained operable throughout the event. The problem was such that it could have existed undetected for some period of time. Because a reactor scram occurred and those systems needed for safe shutdown were challenged, the licensee has committed to revised surveillance testing in an attempt to detect the problem in advance. The loss of power in the auxiliary building which initiated the event was found to be caused by a short circuit.

The NRC Region III conducted an investigation of the event, and the corrective actions taken or planned by the licensee. As stated in a letter to the licensee dated November 5, 1982, no items of noncompliance or deviations associated with the event were identified (Ref. C-1).

Since no plant safety limits were exceeded, the reactor protection system performed as designed, and the licensee responded in a satisfactory manner, there was no impact on public health or safety; therefore, the event is not considered reportable as an abnormal occurrence.

2. Plant Construction Deficiencies at Clinton Nuclear Power Station

On October 5, 1982, the NRC Region III Office (Chicago) proposed a $90,000 civil penalty against Illinois Power Company (the licensee) for violations of the NRC's quality assurance regulations at the licensee's Clinton Nuclear Power Station. The plant, which is under construction in DeWitt County, Illinois, will utilize a boiling water reactor. The violations included inadequate documentation and implementatation of the quality assurance program for electrical work and several instances of alleged intimidation of electrical quality control inspectors by Baldwin Associates, the prime contractor at Clinton (Ref. C-2). The licensee subsequently paid the civil penalty.

The violations were identified in an investigation of allegations made to the NRC's Senior Resident Inspector by several electrical quality control inspectors. The investigation determined that Baldwin Associates had not properly implemented its quality assurance program and that significant construction deficiencies, identified by the contractors's quality control inspectors, were being handled informally rather than by using established procedures for documenting and resolving the deficiencies. As a result of these inspection findings, the licensee issued a stop work order on January 15, 1982, for the installation of electrical cable trays and related activities, including placement of cables in trays which had not been properly inspected.

Subsequently, two quality control inspectors, employed by Baldwin Associates, the constructor, were fired on January 27, 1982. They alleged to NRC personnel that their dismissals were related to statements they had made to the NRC concerning electrical inspection activities. On February 2, 1982, the two inspectors were rehired at the direction of Illinois Power Company.

Additional stop work orders were issued by the licensee on June 23, 1982, covering electrical conduit installation; heating, ventilating, and air conditioning work on safety-related and seismic-related systems; and installation of containment structural steel, electrical equipment, and instrumentation.

The stop work orders were issued by the licensee after the NRC Resident Inspector raised the concern that quality control inspections were lagging far behind the pace of construction. The stop work orders were based on the findings of the licensee and its consultants in addressing the NRC concern.

Quality assurance deficiencies identified by the licensee and by the NRC include inadequate quality and construction procedures, failure to identify construction which did not meet requirements, failure to document construction problems when identified, inadequate training of quality control inspectors, and a significant backlog of quality control inspections to be completed. Actual construction problems found by the licensee and the NRC include placement of electrical cables in cable trays that did not meet requirements, incorrect valve installations, defective welds, incorrect sizing of components, loose parts of fasteners, and missing components.

The licensee is continuing to upgrade and expand its quality control organization for ongoing work and for the resumption of work for which stop work orders have been issued. Strict management controls and quality assurance surveillance are being maintained for what safety-related work is continuing, and Region III personnel have determined that this ongoing work is being performed satisfactorily.

Since the initial stop work in January 1982 there have been a series of management meetings between the licensee and NRC Region III personnel to review the licensee's program for correction of quality assurance and management problems and for resumption of work in those areas where work was stopped.

The licensee retained the firm of Stone and Webster in October 1982 to provide management personnel for certain key positions in the licensee's construction management organizations. There also has been increased involvement with the Clinton project by top licensee management.

The NRC Region III office, through its resident inspector and other regional office personnel, has maintained close surveillance over the activities at Clinton and the plans for resumption of work. A Confirmatory Action Letter was issued to the licensee on January 27, 1982, documenting the licensee's agreement for the actions to be taken prior to resuming the electrical activities (Ref. C-3). A second Confirmatory Action Letter was issued to the licensee on September 1, 1982, documenting the licensee's agreement that it would not lift its stop work orders without NRC concurrence, that it would develop a management plan for continuation of construction and preparation for future plant operation, and that a reinspection and document verification program would be instituted for work completed before June 1982 (Ref. C-4).

Had the plant been fueled and operating, the importance of the deficiencies would have been considerably enhanced. However, since the deficiencies were found while the plant was still under construction, the event is not considered reportable as an abnormal occurrence.

3. NRC Suspension of Safety-Related Construction at Zimmer Nuclear Power Station

In a previous issue of these quarterly reports to Congress (Appendix B, Item 2, of NUREG-0090, Vol. 4, No. 4), it was stated that the NRC issued on November 24, 1981 a Notice of Violation and Notice of Proposed Imposition of Civil Penalties (for $200,000) to the Cincinnati Gas and Electric Company (Ref. C-5), the holder of the Construction Permit for Zimmer Unit 1. The plant, which is under construction in Clermont County, Ohio, will utilize a boiling water reactor. The Notice was for violations of NRC quality assurance regulations which were identified during a ten-month investigation, beginning in January 1981, into delegations of construction deficiencies at the plant site. The licensee paid the civil penalty on February 24, 1982. However, as described below, continuing quality assurance problems resulted in an immediately effective NRC order on November 12, 1982 for the licensee to suspend safety-related construction, including rework activities (Ref. C-6).

Based on preliminary findings during the 1981 ten-month NRC investigation, the licensee had agreed in April 1981 to substantially upgrade its quality assurance program for on-going work. Subsequently, the licensee submitted in August 1981 a Quality Confirmation Program to determine the quality of completed construction work. That quality confirmation program, which is still in progress, identified numerous examples of construction deficiencies, including substandard welds, questionable heat treatment on some small bore piping, electrical cable tray installation and inspection deficiencies, and cable separation problems. In all, the licensee's continuing quality confirmation program has identified approximately 4,200 nonconformances (items which could reflect construction of other types of deficiencies) through December 1982.

NRC inspections, following the issuance of the investigation report in late 1981, continued to identify instances of inadequate quality assurance/quality control activities at the site. In addition the NRC has continued to receive allegations of construction and quality assurance deficiencies at Zimmer; in 1982 the volume of new allegations exceeded the pace of allegations being investigated and completed by the NRC.

An NRC inspection in August and September 1982 identified significant concerns with the implementation of the licensee's quality assurance program and its management program established to control the activities of Catalytic, Inc., a licensee contractor working on control rod drive system hangers and supports, as well as other construction work and rework. These concerns included training of personnel, design control measures, procedure content and implementation, document control, inspection and surveillance activities and other aspects of the contractor's work. On October 11, 1982, the licensee issued a stop work order for the contractor's work.

The licensee had also been proceeding with some rework activities prior to completion of the relevant Quality Confirmation Program tasks. A major example of this rework activity was structural steel welding where approximately 70 percent of structural steel welds were to be reworked to make them acceptable. This rework was initiated before the completion of the Quality Confirmation Program review of all structural steel welds and beam and hanger materials. The rework of the welds involved the addition of new weld material over potentially unacceptable weld material or beam and hanger materials. This approach to rework indicated a lack of a comprehensive management program for rework activities.

In addition to the NRC inspection findings, the National Board of Boiler and Pressure Vessel Inspectors, at the request of the State of Ohio, has been onsite since March 1982 reviewing piping and other work subject to the ASME (American Society of Mechanical Engineers) code. Between March and November 1982, the National Board issued three interim reports documenting deficiencies in the following areas of ASME code work: design control, procurement, procedures, special processes, nonconforming conditions, and corrective actions. These findings are generally consistent with past and present NRC findings.

In view of the importance to safety of construction verification and corrective actions and the past pattern of quality assurance deficiencies, the Commission concluded that safety-related construction, including rework activities, should be suspended until there is reasonable assurance that future construction activities will be appropriately managed to assure that rework activities and all other construction activities will be conducted in accordance with Commission requirements. Therefore, on November 12, 1982, the Commission issued an Order to Show Cause and Order Immediately Suspending Construction, including rework activities, to the licensee (Ref. C-6). The Order, in addition to immediately suspending all safety-related construction at the Zimmer site, requires the licensee to: (1) Obtain an independent review of its management of the Zimmer project by an organization approved by the NRC Regional Administrator to determine measures needed to ensure that construction of the plant can be completed in conformance with the Commission's regulations and construction permit; (2) Submit to the NRC's Regional Administrator for approval the recommended course of action based on the independent management review; (3) Submit to the NRC Regional Administrator for approval an updated comprehensive plan to verify the quality of construction at the Zimmer plant, including provision for an audit of the Quality Verification Program by a qualified outside organization; and (4) Submit to the NRC Regional Administrator a comprehensive plan, based on the results of the verification program, for continuation of construction, including rework activities.

The NRC Region III Office is closely monitoring the actions being taken and planned by the licensee in response to the Commission Order. Management meetings and inspection efforts will be scheduled as necessary.

Since the deficiencies were found while the plant was still under construction, the event is not considered reportable as an abnormal occurrence. The safety significance would have been considerably enhanced had the numerous deficiencies been discovered after the plant was fueled and operating.

4. Medical Misadministration

On April 16, 1982, an official of the Medical College of Ohio at Toledo notified the NRC Region III office that a female cancer patient had received a radiation exposure substantially greater than prescribed. The patient was a middle-aged female with a diagnosis of cancer of the uterus. The prescribed treatment plan consisted of external beam radiotherapy, followed by vaginal radioisotope application with subsequent hysterectomy. The external radiation therapy was completed on April 13, 1982. The second phase of the treatment began on April 12, 1982. Four cesium-137 sources, loaded in a tube, were placed in the patient. The sources were removed on April 16, 1982. The prescribed source loading and time of application were chosen to deliver a calculated dose of 4,000 rad.

When the tube was removed, it was turned over to a radiation physicist. Shortly after, the radiation physicist unloaded the tube and discovered that three of the sources were each approximately three times the intended strength. The fourth was the correct strength. This resulted in the patient receiving approximately 12,000 rad rather than the intended 4,000 rad to the localized area.

The misadministration was later discussed with the patient's referring physician who agreed that the patient should be informed. Accordingly, during the patient's first post-treatment followup visit on April 19, 1982, the misadministration was discussed with the patient and her husband. At this time, the patient had not yet experienced any untoward effects of this treatment. The patient was admitted to the hospital on July 13, 1982, for the third phase of her original treatment plan and was discharged from the hospital on July 22, 1982.

On August 2, 1982, the patient experienced the abrupt onset of radiation-induced complications. The patient was treated and some of the complications were diminished. The long-term prognosis is difficult to assess. The degree of radiation intolerance varies in individual patients; in addition, specific tissues vary in their tolerance to radiation. If the complications do not heal, further medical treatment may be prescribed.

The cause of the misadministration was due to an error made by a registered, radiation therapy technologist who was being trained as a dosimetrist. The hospital's procedures regarding the use of the sources included a source record form. The various sections of the form are filled out by the individual who performed the specified action as sources are moved from the storage safe, to the patient, and back to the storage safe. Even though the prescribed loading was listed on the source record form, and even though the hospital's sources are color coded as to their strength, the technologist incorrectly loaded the tube with three sources out of the four having strengths over three times that prescribed. The techologist was experienced in removing and returning sources from or to the storage safe and was familiar with the source color coding system; he could offer no reason for choosing the wrong sources.

The licensee informed the NRC Region III office of the misadministration on April 16, 1982. This was followed by a written report on April 28, 1982. The report included the details of the event and the corrective actions taken.

The licensee stated that procedures for handling the sources were being revised. Also, the source record form was being changed to emphasize the verification of the correct source loading by physicians prior to administering the application to patients.

The radiation oncologist, physicist and dosimetrist were instructed that all source loadings are to be verified by someone other than the individual who loads the sources prior to their administration. The radiation oncologists were instructed that their signature on the source record form in the certificate of receipt section signifies that they have verified and acknowledged receipt of the prescribed sources. To act as an additional safety check, the exposure rate levels at one meter for various source loadings of the different applicators would be established. The measured exposure rate levels at one meter for each patient would then be compared with these expected levels. If the comparison is not within plus or minus 20%, the safe will be rechecked to ensure that the appropriate sources have been removed from the safe drawer. If the comparison is not within plus or minus 50% of the expected levels, the sources will be removed from the applicator and the loading checked.

The NRC Region III staff performed a special inspection on April 19, 20, and 23, 1982, which consisted of a selective examination of procedures and representative records, observations, independent measurements, and interviews with personnel. Also, the circumstances surrounding the specific misadministration were reviewed. This was followed by a management meeting between the hospital and Region III personnel on May 27, 1982. Based on discussions at this meeting, the licensee planned to request a license amendment to incorporate corrective actions. The amendment was subsequently submitted and incorporated into the license.

In a NRC Region III letter to the licensee on July 15, 1982, the licensee was cited for two violations, neither of which were contributing factors in the misadministration; i.e., (a) failing to perform a survey of radiation levels in unrestricted areas adjacent to an implant patient room on April 12, 1982, and (b) not including some sources in a physical inventory conducted on October 19, 1981. The licensee responded to the violations on August 10, 1982, including revised procedures to prevent recurrence. The NRC Region III staff will further examine the corrective actions taken by the licensee at some future inspection.

The NRC medical consultant was requested by NRC Region III to review the misadministration. His report was submitted to NRC Region III on September 13, 1982.

REFERENCES
(FOR APPENDICES)

B-1 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 82-03, "Steam Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants," October 14, 1982.*

B-2 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 82-03, Revision 1, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants," October 28, 1982.*

C-1 Letter from R. L. Spessard, Director, Division of Projects and Resident Programs, NRC Region III, to Cordell Reed, Vice President, Commonwealth Edison Company, forwarding Inspection Reports No. 50-295/82-22 and No. 50-304/82-19, Docket Nos. 50-295 and 50-304, November 5, 1982.*

C-2 Letter from J. G. Keppler, Regional Administrator, NRC Region III, to W. C. Gerstner, Executive Vice President, Illinois Power Company, transmitting a Notice of Violation and Proposed Imposition of Civil Penalties, Docket No. 50-461, October 5, 1982.*

C-3 Confirmation of Action Letter from J. G. Keppler, Regional Administrator, NRC Region III, to W. C. Gerstner, Executive Vice President, Illinois Power Company, Docket No. 50-461, January 27, 1982.*

C-4 Confirmatory Action Letter from J. G. Keppler, Regional Administrator, NRC Region III, to W. C. Gerstner, Executive Vice President, Illinois Power Company, Docket No. 50-461, September 1, 1982.*

C-5 Letter from R. C. DeYoung, Director, NRC Office of Inspection and Enforcement, to W. H. Dickoner, President, Cincinnati Gas and Electric Company, forwarding a Notice of Violation and Notice of Proposed Imposition of Civil Penalties, Docket No. 50-358, November 24, 1981.*

C-6 U.S. Nuclear Regulatory Commission Order to Show Cause and Order Immediately Suspending Construction, signed for the Commission by J. C. Hoyle, Acting Secretary of the Commission, issued in the matter of Cincinnati Gas and Electric Company (licensee for William H. Zimmer Nuclear Power Station), Docket No. 50-358, November 12, 1982.*

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

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0090, Vol. 5, No. 4
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	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH May YEAR 1983
9. 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	DATE REPORT ISSUED MONTH May YEAR 1983
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16. 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 or safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 to December 31, 1982.

During the report period, there was one abnormal occurrence at the NRC licensees. The event involved the containment spray system being inoperable at one of the nuclear power plants licensed to operate. The Agreement States reported no abnormal occurrences to the NRC.

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

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

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