NUREG-0090
Vol. 7, No. 1

Report to Congress on Abnormal Occurrences

January - March 1984

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

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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

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 January 1 to March 31, 1984.

The report states that for this report period, there were three abnormal occurrences at the nuclear power plants licensed by the NRC to operate. The first involved an inoperable containment spray system; the second involved a through wall crack in a vent header inside a BWR containment torus; and the third involved a serious degradation of a reactor depressurization system. There were two abnormal occurrences for the other NRC licensees. The first involved an overexposure to a member of the public; and the second involved a therapeutic medical misadministration. There was one abnormal occurrence reported by the Agreement States; the event involved an overexposure of a radiographer and assistant.

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 between January 1 to March 31, 1984.

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

THE REGULATORY SYSTEM

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

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

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

REPORTABLE OCCURRENCES

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

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

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

Some form of LER reporting system has been in existence since the first nuclear power plant was licensed. Reporting requirements were delineated in the Code of Federal Regulations (10 CFR), in the licensees' technical specifications, and/or in license provisions. In order to more effectively collect, collate, store, retrieve, and evaluate the information concerning reportable events, the Atomic Energy Commission (the predecessor of the NRC) established in 1973 a computer-based data file, with data extracted from licensee reports dating from 1969. Periodically, changes were made to improve both the effectiveness of data processing and the quality of reports required to be submitted by the licensees.

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

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

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

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

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

AGREEMENT STATES

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

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

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

FOREIGN INFORMATION

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

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

JANUARY-MARCH 1984

NUCLEAR POWER PLANTS

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

84-1 Inoperable Containment Spray System

The following information pertaining to this event is also being reported concurrently in the Federal Register. 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 November 29, 1983, Consolidated Edison of New York (the licensee) discovered that two motor operated spray header discharge valves at Indian Point Unit 2 were found in the locked-closed, de-energized position instead of the required locked-open, de-energized position. This condition would have prevented automatic actuation of the containment spray system during the safety injection phase of an accident. Indian Point Unit 2 utilizes a Westinghouse-designed pressurized water reactor and is located in Westchester County, New York.

Nature and Probable Consequences - During a cold shutdown for unscheduled plant maintenance, the spray header discharge valves (MOVs 869A and 869B) were closed and tagged out of service. Following the maintenance, personnel were assigned to perform a check-off procedure which should have returned the valves to their proper position prior to heating the reactor coolant system above 350°F and subsequent core criticality. However, due to personnel errors in completing the check-off procedure, this was not done.

On October 25, 1983, the licensee completed the unscheduled maintenance and returned the reactor to criticality. Four reactor trips occurred during the plant startup period. The unit was returned to full power operation on October 28, 1983. The unit operated at or near 100% power through November 22, 1983 when the reactor automatically shut down due to an equipment problem. Repairs were made and power operation resumed on November 25, 1983. On November 29, 1983, with the reactor operating at 100% power, the licensee discovered that MOV 869A and MOV 869B were closed, while performing a bimonthly (every two months) containment spray pump surveillance test.

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 (LOCAs) and to reduce the pressure and temperature to minimize containment leakage. The system 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 trains and fan coolers that would operate following such an accident. The containment fans, in conjunction with a filtration system, would also remove some radioactive iodine in the post-accident containment atmosphere; however, this method is not as effective as the containment spray system.

The containment heat removal system consists of five containment fan cooler units and two containment spray trains. The plant's final safety analysis report (FSAR) states that sufficient post-accident heat removal capability can be provided by any of the following combinations:

1. All five containment fan cooling units;

2. Both containment spray trains (and one of the two recirculation spray trains during the recirculation phase of safety injection); or

3. Three containment fan cooler units and one containment spray train.

During the time in question, automatic actuation of the containment spray system would not have been possible. However, there are indications in the control room which could inform the reactor operators that spray injection was not taking place. The operators then have various options to manually initiate containment spray, e.g. (1) realign the spray valves from the motor control center, an area designed to be accessible in high, post-accident radiation fields, or (2) supply spray from the residual heat removal discharge by opening appropriate valves from the central control room.

Although the reactor operators would be expected to recognize in a timely manner that the containment spray valves were closed, the NRC staff has performed bounding calculations to predict worse case conditions in order to determine whether either the containment design pressure or post-accident off-site dose limitations would be exceeded after a design basis accident. Indian Point Unit 2 has two trains of fan coolers on separate power sources; one train has two fan coolers and the other train has three fan coolers. Since, for the present situation, both containment spray trains would be out of service, the staff assumed that a single active failure would reduce the active containment heat removal capability to two fan coolers during a pipe break accident. Under these conditions, the reduced heat removal capability would be expected to result in a higher peak containment pressure. In addition, less filtration of radioactive iodine would be expected to result in higher off-site doses.

The NRC calculations show a peak containment pressure, for the design basis loss of coolant accident (double-ended pump suction guillotine break), of 41.9 psig; this is substantially below the containment design pressure of 47 psig. However, based on the methods and assumptions consistent with those in the current licensee application reviews (i.e., Standard Review Plan 15.6.5), calculations predict resultant doses approximately four times the 10 CFR Part 100 thyroid exposure guidelines at the exclusion area boundary, assuming no operator action. If operator action were to be taken to initiate containment spray after 30 minutes, calculations predict resultant doses approximately 1.8 times the exposure guidelines at the exclusion area boundary.

These calculations are expected to be very conservative. Possible mitigating factors are:

1. The calculations assume the worst case single active failure (i.e., the power source that powers three of the five containment fan cooler units). In addition, credit is not given to operator action to actuate the containment spray systems prior to 30 minutes.

2. The dose calculations assumed the standard containment leak rate of 0.1% for the first 24 hours. Credit for a reduced leak rate was not given for either (1) the actual, as measured, containment leak rate or (2) the Isolation Valve Seal Water System which automatically injects water between the containment isolation valves post-accident in order to eliminate potential containment leak paths.

However, it should be noted that in regard to Item 1 above, even if the worst case single active failure is <u>not</u> assumed (i.e., all five containment fan coolers are operating), NRC calculations predict iodine doses at the exclusion area boundary which exceed the 10 CFR Part 100 guidelines.

<u>Cause or Causes</u> - The cause of the event is attributed to personnel error. On October 23 and 24, 1983, prior to plant startup after the maintenance outage, operators were assigned to perform a Safety Injection System Check-Off List (COL-12) which should have returned the valves to their proper positions. COL-12 required one operator to ensure the correct valve position and a second operator to verify the position. COL-12 directs the operators to the motor control centers to perform two verifications for each valve: (1) verify that the position of the valve is open, and (2) verify that the breaker is de-energized. In the de-energized condition, position indication for the valve is lost at the motor control centers. Verifying position at the motor control center, therefore, requires energizing the breaker. This was not done, and each operator assumed the valve was open. The first operator assumed that the valve was positioned by another operator. The second operator assumed the valve was open because the breaker was locked in the de-energized position.

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

<u>Licensee</u> - On November 29, 1983, while performing a routine containment spray surveillance test, test personnel realized the valve line-up was wrong when the "as left" position differed from the "as found" position. The senior reactor operator was notified when the discrepancy was identified and the valves were positioned correctly.

The licensee reported the incident to the NRC Resident Inspector and by telephone to the NRC Operations Center. The licensee initiated an investigation to establish the cause of the event and to determine corrective actions. The investigation included interviews with cognizant personnel and review of pertinent procedures, qualification programs, technical specifications, and other reference documentation. Immediate corrective action steps taken by the licensee included verifying correct valve positions of similarly de-energized safeguards valves found on check-off lists.

In addition, the licensee determined that improvements could be made in the training/qualification program of nuclear plant operators to place new emphasis on equipment status identification. The operator qualification standard will specify the knowledge required by the operator for the performance of COLs. In addition, the licensee will further assure that appropriate guidance is provided to the operators in the conduct of COLs.

Other long term corrective actions include: (1) review of valve position indication for all safety related valves to determine if modifications are necessary to provide for positive indication of de-energized valves, and (2) verification of the operability of all currently installed safety related MOV position indicators with corrections if necessary.

NRC - An investigation of the details associated with the event was made as part of the routine inspections conducted by the Resident Inspectors at the plant during the period from October 18 to November 30, 1983. One violation was noted, i.e., failure to meet a technical specification Limiting Condition for Operation with respect to the operability of the containment spray system.

On December 13, 1983, an enforcement conference was held between NRC Region I personnel and the licensee. The safety significance and immediate and long-term corrective actions for the event were discussed.

On March 13, 1984, the NRC Region I forwarded a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $40,000. In addition, the NRC will monitor the actions taken by the licensee to prevent recurrence.

The NRC notes that there have been several events at various nuclear power plants which involved inadvertent isolation of either the containment spray system or the chemical (sodium hydroxide) addition tanks while the plants were at power. These events are briefly described in Table 1. While most of the events only resulted in system inoperability for a few minutes or hours, the potential was there for extended plant operation with these safety systems inoperable.

Three events in Table 1 caused particular concern because of the extended periods of time in which the containment spray systems were inoperable. The first was the October 28, 1982, event at Farley Unit 2 in which the systems were inoperable for over 17 months. This event was reported as abnormal occurence 82-7 in NUREG-0090, Vol. 5, No. 4 ("Report to Congress on Abnormal Occurrences: October-December 1982").

Table 1

Events Involving Inadvertent Isolation of Containment Spray Systems

Plant name; Licensee; Plant Location; Date of Event	Event
Davis Besse; Toledo Edison Co.; Ottawa County, OH; January 12, 1978	Both containment spray pumps found with the circuit breakers de-energized. Personnel error resulted in 24 hours of plant operation (Mode 4) with system inoperable.
Davis Besse; Toledo Edison Co.; Ottawa County, OH; December 28, 1978	Two hours after entering Mode 4, containment spray pump motor breakers were found in the lockout position. Caused by failure to follow procedures.
D.C. Cook Unit 2; Indiana & Michigan Electric Co.; Berrien County, MI; May 2, 1978.	During change from Mode 5 to Mode 4, containment spray pumps remained inoperable when control switches were left in lockout position. Procedural and personnel error left system inoperable for four hours.
Farley Unit 1; Alabama Power Co.; Houston County, AL; May 10, 1982.	While performing "Penetration Room Exhaust and Air Filtration System Train Operability and Valve Inservice Test," an operator inadvertently closed the containment spray suction valves from the refueling water storage tank. The valves were closed at power for seven hours.
Farley Unit 2; Alabama Power Co.; Houston County, AL; December 26, 1981.	While performing surveillance testing, the isolation valve on the NaOH spray additive tank was found in the closed position. Caused by operator error during position alignment checks.
Farley Unit 2; Alabama Power Co.; Houston County, AL; October 28, 1982.	The containment spray header isolation valves on each of the two supply headers were found locked in the closed position. Condition had existed for over 17 months. Caused by valves not being in conformance with design drawings and by a procedural inadequacy.
Ginna; Rochester Gas & Electric Corp.; Wayne County, NY; June 13, 1983.	While changing modes (cold shutdown to hot shutdown), the containment spray pumps were found in the pull-to-lock position.

Table 1 (continued)

Events Involving Inadvertent Isolation of Containment Spray Systems

Plant name; Licensee; Plant Location; Date of Event	Event
Indian Point Unit 2; Consolidated Edison Co. of New York; Westchester County, NY; September 24, 1980.	Both containment spray pump control switches found in pull-to-lock position by the NRC resident inspector while the plant was at full power. Plant procedures called for such practices during containment entry. Licensee informed of non-compliance with Technical Specifications and procedures subsequently revised.
Indian Point Unit 2; Consolidated Edison Co. of New York; Westchester County, NY; November 29, 1983.	While performing a containment spray pump surveillance test, during normal operation, two motor operated spray header isolation valves were found in the locked closed, de-energized position instead of the required locked open, de-energized position. Condition had existed for about five weeks. Caused by personnel error.
McGuire Unit 1; Duke Power Co.; Mecklenburg County, NC September 29, 1983.	Both trains of containment spray system were technically inoperable for about five hours while the plant was operating at full power. The cause was a combination of component failure and operator error.
Point Beach Unit 1; Wisconsin Electric Power Co.; Manitowoc County, WI; June 21, 1981.	While performing periodic surveillance, the spray additive tank isolation valve was found in the closed position, thus preventing injection of NaOH to the containment spray system. Operator error leaves valve misaligned for four days.
San Onofre Unit 3; Southern California Edison; San Diego County, CA; March 17, 1984.	While performing routine surveillance at nearly full power, manual isolation valves in both of the containment spray headers were found closed. System was inoperable for about 13 days. Cause of the misalignment of the isolation valves was improper use of the valve alignment checklist.
Surry Unit 1; Virginia Electric & Power Co.; Surry County, VA; October 16, 1982.	Isolation valves leading from the chemical addition tank were found in the closed position. Cause attributed to personnel failure to perform valve alignment check.
Turkey Point Unit 4; Florida Power & Light Co.; Dade County, FL; October 4, 1983.	A non-licensed operator assigned to close the spray header isolation valves on Unit 3 (cold shutdown) inadvertently closed the identical valves on Unit 4. Unit 4 operated at power with these valves closed for 50½ hours.

The second was the November 29, 1983, event at Indian Point Unit 2 in which the systems were inoperable for about five weeks. This event is discussed above as an abnormal occurrence.

The third is the March 17, 1984, event at San Onofre Unit 3 in which the systems were inoperable for about 13 days. This event is still under evaluation. If it is determined to meet the abnormal occurrence reporting threshold, it will be included in a succeeding issue of these quarterly reports to Congress.

On May 25, 1984, the NRC issued Inspection and Enforcement Information Notice No. 84-39 (Ref. 1) to all facilities holding an operating license or construction permit, which was based on information contained in Table 1. This may help to reduce the frequency of these types of events by heightening the industry's awareness of the potential for such events and the circumstances associated with their occurrence.

This incident is closed for purposes of this report.

* * * * * *

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

The following information was previously reported in the Federal Register (Ref. 2). Appendix A (see general criterion 2) of this report notes that major degradation of essential safety-related equipment can be considered an abnormal occurrence.

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

Nature and Probable Consequences - The primary containment for Hatch Unit 2 is a pressure suppression system consisting of a drywell, pressure suppression chamber (torus) containing a large volume of water, a connecting vent system between the drywell and water pool, isolation valves, vacuum relief systems, containment cooling systems, and other equipment. The drywell is a steel pressure vessel which houses the reactor vessel, the recirculation system, and other systems and components important to safety.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus which is located below the drywell. Eight circular vent pipes form a connection between the drywell and the torus. The vent pipes exhaust into the 54-inch diameter continuous vent header, which is located in the torus, from which 80 downcomer pipes extend downwards into the water in the torus.

During operation, the drywell and suppression chamber free air spaces are inerted with nitrogen to minimize the possibility of hydrogen combustion during or following a loss of coolant accident (LOCA). The nitrogen supply system is designed to evaporate liquid nitrogen and warm the nitrogen gas before it is discharged into the primary containment. At Hatch, the temperature of the nitrogen gas at the vaporizer controller is normally controlled between 100-250°F. By the time the gas reaches the discharge outlet to the torus, the gas temperature would be somewhat lower. However, under worst case conditions of equipment failure, the discharge temperatures into the torus could drop far below 0°F. Such temperatures would cool materials below their nil ductility temperatures causing them to become susceptible to brittle fracture.

Hatch Unit 2 was shut down on January 13, 1984, for an extended outage to replace recirculation piping. On February 3, 1984, during a routine visual inspection of the torus interior, the licensee discovered the circumferential crack in the 54-inch diameter torus vent header. The ends of the pipe on either side of the crack were displaced about 1/2 inch. Further inspection showed that the through wall crack extended about 330 degrees around the header. The vent header has a wall thickness of 0.25 inch.

The containment system is designed such that in the event of a LOCA, pressurized steam and water is released into the drywell. Drywell pressure quickly increases and forces the steam flow through the vents into the vent header. The vent header directs the steam through the downcomer pipes into the torus water resulting in condensation of the steam. The condensation of steam serves to limit the maximum pressure the containment structure will experience. However, as a result of the large through wall crack in the vent header, the amount of steam condensed by the torus would be reduced because some steam would bypass the vent header and reduce the differential in pressure used to drive the steam into the water. This increases the possibility of overpressurizing the primary containment, allowing for a release into the secondary containment. This condition has not been specifically analyzed in the plant's final safety analysis report (FSAR), thus leading to a serious safety concern.

Cause or Causes - The location of the crack was directly below a nitrogen discharge outlet to the torus. The nitrogen line is 20 inches in diameter with the outlet about seven feet above the vent header. The licensee stated that there have been problems with operation of the nitrogen evaporators and heaters, and that the low temperature isolation provisions had also malfunctioned. The licensee reported that the nitrogen inerting system had recently (at least) been used without the gas pre-warmer equipment working because of failure of the site auxiliary boiler. In addition, several nitrogen system valves were reported to have failed because of frost/ice buildup. Component failures, combined with deficient management/procedural controls pertaining to containment inerting evolutions, are believed to be the principal causes of the cracking problem.

The crack was determined to be a brittle-fracture type of failure. The primary contributor to the cracking was attributed to impingement of low temperature nitrogen onto the vent header. Apparently, the vent header temperature dropped below the nil ductility temperature when the nitrogen evaporator and heater and/or isolation system were not functioning properly. The thermal stresses generated by this cooling contributed to crack initiation and propagation. The vent header material for Hatch Unit 2 is SA 516 Grade 70 carbon steel with the nil ductility transition temperature below -40°F.

Actions Taken to Prevent Recurrence

Georgia Power Company - The licensee performed operational testing of the vent header on February 3, 1984 on its operating reactor, Hatch Unit 1, and shut down the unit on February 4, 1984, to further verify that the same condition did not exist. Visual inspections of the vent header were made and no cracks were found. Unit 1 was then restarted. It was also determined that the Unit 1 nitrogen line discharge was not located directly above the vent header. For Unit 2, repairs to the vent header have commenced. Long range actions, similar to those described below, will also be required.

The licensee removed samples from the vent header for failure analysis examination by General Electric (GE). This analysis confirmed that the material met all physical and chemical requirements for SA 516 Grade 70 carbon steel and that the failure mode was transcrystalline brittle fracture characteristic of crack propagation at temperatures below the nil ductility transition temperature.

Vendor/Other Licensees - In response to the NRC Bulletin described below, the licensees for Pilgrim, Oyster Creek, Dresden Unit 3, Quad Cities Unit 2, and Brown's Ferry Unit 3 visually inspected their vent headers for cracking. No indications of cracking were found. In addition, the licensee for Monticello, which had shut down after the Bulletin was issued, voluntarily performed visual inspections; again, no indications were found.

The BWR Regulatory Response Group (RRG) was activated. This group, together with the vendor (GE) representatives met with the NRC on February 6 and 23, 1984 to discuss actions to be taken to assure the integrity of the containment and associated systems, and to determine whether any design and/or procedure changes are necessary. This work is continuing.

General Electric issued a service information letter (SIL) which contains recommended actions to be taken by all BWR owners with Mark I or Mark II containment systems (Ref. 3). The actions involve evaluations of inerting system design and operation, performance of a leakage test to confirm the integrity of the vent system, inspection of the nitrogen injection line, and inspection of containment components and equipment. The owners' group letter transmitting the SIL requested that the licensees and applicants report their findings to the NRC.

NRC - An NRC team was dispatched to the site on February 4, 1984, to participate in the investigation of the event. As part of the investigation, sample material from the failed vent header was obtained by NRC Region II for independent failure analysis by Brookhaven National Laboratory. The work done by Brookhaven confirmed that the material met SA 516 Grade 70 physical and chemical requirements and that the failure mode was characteristic of crack propagation at temperatures below the nil ductility transition temperature; these results are consistent to those obtained by the licensee's vendor, GE.

On March 14, 1984, NRC Region II forwarded to the licensee a notice of violations based on inspections performed at Hatch Units 1 and 2 between January 21 and February 20, 1984 (Ref. 4). The violation germane to the vent header problem pertained to procedural inadequacies in not properly implementing procedure HNP-2-1500, Primary Containment Atmospheric Control Systems. The procedural inadequacy allowed the nitrogen temperature at the vaporizer controller to drop, during containment inerting, below the specified band of 100-250°F. The procedure specifically cautioned against operation of the vaporizer below 100°F, but did not specify actions to be taken if the temperatures did fall below the specified band.

Inspection and Enforcement (IE) Bulletin No. 84-01, dated February 3, 1984, was issued to all BWRs with operating licenses or construction permits (Ref. 5). The IE Bulletin requested that facilities with operating licenses in cold shutdown and with primary containments similar to the Hatch containment (Mark I) perform inspections as to the condition of their vent headers and report the results to NRC. It was also recommended that, for BWRs with Mark I containments that were in operation, the licensees review plant data on differential pressure between the drywell and the torus for anomalies that could be indicative of cracks. The results of these initiatives are under review.

On March 5, 1983, IE Information Notice No. 84-17 was sent to all reactor facilities with operating licenses or construction permits to alert them to possible problems associated with cooling components to below their nil ductility temperatures with liquid nitrogen (Ref. 6). The Notice also advised licensees and applicants of potentially similar problems associated with the use of other very cold fluids where the fluid could come in contact with safety-related components subject to brittle fracture.

The NRC has met with the vendor and the BWR RRG to determine whether the problem is unique to Hatch Unit 2, and whether other actions need to be taken to prevent recurrence of the problem. All aspects relevant to the failure will be reviewed in addition to the repairs made to Hatch Unit 2.

The NRC staff will review the licensees' responses to the recommendations in the General Electric SIL, and determine if there is a need for further actions.

Future reports will be made as appropriate.

* * * * * *

84-3 Serious Degradation of Reactor Depressurization System

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see general criterion 2) of this report notes that major degradation of essential safety-related equipment can be considered an abnormal occurrence.

Date and Place - On February 22, 1984, the NRC was notified by Consumers Power Company, licensee for the Big Rock Point Nuclear Power Station, that three of four reactor depressurization system (RDS) isolation valves failed to open during a surveillance test at 1:15 a.m. This surveillance testing is routine testing which is performed every 90 days. Big Rock Point is a boiling water reactor located in Charlevoix County, Michigan.

Nature and Probable Consequences - At the time of the event, the plant was in hot standby condition (reactor shut down, system at reduced pressure and temperature - approximately 50 psig and 265°F, respectively). The plant had been shut down since February 19, 1984, for various maintenance activities. When the three isolation valves failed to open during the surveillance test, the licensee declared the incident to be an Unusual Event (the least severe of the NRC's emergency categories) until the plant was placed in cold shutdown (reactor shut down, system at atmospheric pressure and temperature below 212°F).

The RDS is a set of piping and valves which was installed at Big Rock Point in the mid-1970's. One large pipe from the steam drum feeds four parallel lines; each line contains an isolation valve and a depressurization valve (both normally closed). Both valves must open to allow flow through the line. The purpose of the RDS is to provide a method of rapidly depressurizing the reactor in the event of a small break loss of coolant accident (SB-LOCA). In such an accident the reactor would lose cooling water while the system pressure would remain high. Since Big Rock Point does not have a high pressure injection system, the RDS reduces the system pressure to the point (roughly 75 psig) where the core spray system (a low pressure system) can deliver cooling water to the reactor. The plant technical specifications require that three of the four lines be operable whenever the reactor is not in cold shutdown. Safety analysis calculations indicate that three lines would be needed to properly depressurize the reactor under the worst case accident conditions. If the RDS did not operate properly in the event of a SB-LOCA, use of the core spray system could be delayed and the core could become uncovered and damaged.

The isolation valves are 6-inch flexible wedge-type gate valves manufactured by Anchor-Darling. The valves are opened by a spring and closed by a pressurized air system. In 1983 the licensee installed an air amplifier system to increase the air pressure which holds the valves closed. No change was made to the springs.

Cause or Causes - After consulting with the valve manufacturer and conducting tests of the valves, the licensee determined that the cause of the valves' failing to open was a combination of thermal binding and the increased force holding the valves closed due to the recently installed air amplifier system. Thermal binding occurs when the valve is closed hot and then cooled down. The cooling causes contraction of the valve seat and therefore requires additional force to open the valve. The increased force holding the valve closed resulting from the installation of the air amplifier further heightened the effects of thermal binding to the point that the springs were not strong enough to open the valves.

Based on the results of past testing, the licensee concluded that the valves would have opened at normal operating temperature which is approximately 550°F. Since the valves failed to open at approximately 265°F and there was no testing at temperatures between 550°F and 265°F, the licensee was unable to determine the temperature at which failures would have begun.

In reviewing past operating experience, the licensee determined that prior to the installation of the air amplifier, there had been no instances of valves failing to open because of thermal binding.

Actions Taken To Prevent Recurrence

Licensee - The licensee removed the air amplifier system from service, and returned to the closing air pressure used previously. This action reduced the force holding the valve closed and minimized the potential for thermal binding. The licensee disassembled one valve for inspection with no defects found. The valves were then cycled at operating temperature and retested during a partial unit cooldown and depressurization. All valves functioned properly during these tests. The licensee also committed to test the valves again during the next cold shutdown.

NRC - The NRC's Senior Resident Inspector arrived at the site at 3 a.m., February 22, 1984. He remained on site until the plant was in cold shutdown. He then monitored the licensee's activities in investigating the cause of the failures and developing corrective actions.

On March 3, 1984, NRC Region III (Chicago) issued a Confirmatory Action Letter confirming the licensee's commitments in testing and examining the valves before returning the plant to operation (Ref. 7). The Senior Resident Inspector witnessed the testing activities.

Having satisfactorily completed the testing and inspections required by the Confirmatory Action Letter, the licensee was given permission to resume normal operations.

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 first calendar quarter of 1984. 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, 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 first calendar quarter of 1984. As of the date of this report, the NRC had determined that the following were abnormal occurrences.

84-4 Overexposure to a Member of the Public

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 2 of "For All Licensees") of this report notes that an exposure to an individual in an unrestricted area, such that the whole body dose received exceeds 0.5 rem in one calendar year, can be considered an abnormal occurrence.

Date and Place - On December 30, 1983, a representative of the University of Cincinnati Hospital, of Cincinnati, Ohio, reported that a radiation therapy device had been handled by hospital personnel who believed it to be empty when, in fact, it actually contained some tiny, sealed, iridium-192 radiation sources.

Nature and Probable Consequences - The radiation therapy device consisted of a plastic template and a series of hollow needles. The device had been borrowed from the University of Cincinnati Hospital by Bethesda Hospital (also of Cincinnati, Ohio) for use in treating a patient. At Bethesda Hospital, the template and needles were surgically fitted to the patient and an x-ray taken to verify that the device was properly positioned. The x-ray also showed that the needles did not contain any radiation sources.

The radiation sources (called seeds), containing iridium-192 and encased in plastic ribbons, were then inserted into 32 of the 42 needles. According to the physician, the ribbons were removed when the treatment was completed on November 23, 1983, and a radiation survey was performed to assure that all had been removed. The treatment device was then removed and cleaned. Hospital personnel who performed the cleaning stated that there were no ribbons or seeds remaining in the needles.

The device was then stored until about December 2, 1983, when it was taken by a secretary to be returned to the University of Cincinnati Hospital. It remained in the secretary's automobile until she gave it to another person to return. After the device was received by the University of Cincinnati Hospital, it was unintentially returned by mail to the treating physician. It was then returned, finally arriving at the University of Cincinnati Hospital on about December 16, 1983.

The device was placed in storage. On December 19 and again on December 26, 1983, it was taken out of storage and used in treatment planning. When not used in planning or placed in storage, the device was left at a receptionist's desk at the hospital for a total of about 4 and 1/2 days. On December 28, 1983, during preparations for a radiation therapy procedure, University of Cincinnati Hospital personnel found a strand of nylon ribbon containing nine seeds in one of the needles. The other needles were checked on December 29, 1983, and two of them were found to also contain ribbons--one with two seeds and one with seven seeds.

About 50 persons at the two licensees received radiation exposures as a result of the incident, according to information gathered by NRC inspectors through interviews with personnel at the two institutions. One University of Cincinnati Hospital employee, an administrative worker who is considered a member of the public and not a radiation worker, received a whole body exposure estimated to be between 750 and 800 millirems. A second administrative employee received a whole body exposure estimated to be 400 to 500 millirems and, in addition, received an exposure of between 15 to 18 rems to the hands.

The other individuals received lesser exposures with most of the exposures being less than 50 millirems. These exposures are estimated from interviews with the individuals involved, since as administrative employees who do not normally handle radioactive materials, they were not wearing radiation measuring devices.

NRC regulations provide that licensed activities should not result in a whole body exposure of a member of the public of more than 500 millirems in any one year. Individuals classified as radiation workers may receive up to 1,250 millirems (1.25 rems) in a calendar quarter and 18.75 rems to the hands.

The exposures received by all of the personnel involved are small and no clinically detectable effects would be expected. However, they do represent unnecessary exposures. The NRC considers that all unnecessary radiation exposures should be avoided as a matter of prudence.

Cause or Causes - The cause of the incident could not be determined with certainty. The physician at Bethesda Hospital stated that all required radiation surveys were performed after the sources were removed from the patient. The NRC requires that surveys be performed of the patient and of areas where the sources were put in place and removed. The physician reported that the surveys showed no evidence of any sources remaining, although the surveys were not documented as required by NRC regulations.

No procedures were in existence for the checking of radiation therapy devices transferred between hospitals, and therefore the device was not surveyed when it was received at the University of Cincinnati Hospital.

Actions Taken to Prevent Recurrence

Licensees - Each licensee was required by the NRC to develop procedures to ensure that all radiation sources are removed from therapy devices and to check equipment being transferred between hospitals. These procedural modifications were made.

NRC - The NRC, through its inspections, was unable to determine responsibility for the mishandling of the sources and subsequent unnecessary radiation exposures. The programs for the control of potentially radioactive materials at both hospitals were found to need improvement. Therefore, each hospital was required to submit its planned actions to improve its handling procedures to prevent a recurrence of this type of incident. In addition, a Notice of Violation was issued to Bethesda Hospital for violations of NRC requirements, including the failure to maintain records of radiation surveys performed after removal of sources from the patient.

This incident is closed for purposes of this report.

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84-5 Therapeutic Medical Misadministration

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 a major reduction in the degree of protection of the public health or safety can be considered an abnormal occurrence.

Date and Place - On March 6, 1984, a representative of Henry Ford Hospital, Detroit, Michigan, reported that a 26-year-old female patient had received a therapeutic radiation dose to the head which was 45 per cent in excess of that prescribed. The misadministration had occurred in a radiation treatment program which began January 30, 1984, and was terminated on March 5, 1984, when the excessive radiation dose was discovered.

Nature and Probable Consequences - Following surgery for a malignant brain tumor, radiation therapy was prescribed for the patient. The treatment plan called for 30 treatments of 200 rads of radiation for a total radiation dose to the midline of the brain of 6,000 rads. (A rad is a standard measure of radiation.)

The normal procedure is to administer half of the radiation dose (or, in this case, 100 rads) to each side of the head. The dosimetrist (the hospital employee who calculates the exposure time necessary to achieve the prescribed dose) erred in calculating the exposure time so that 200 rads was administered to each side of the head--twice the intended amount of radiation per treatment.

The treatment series began January 30, 1984. The patient developed erythema (severe reddening of the skin) during the treatment course. Because this condition was more severe than anticipated, the attending physician reduced the per-treatment prescribed dose to 150 rads after the 15th treatment. A second dosimetrist calculated the new exposure time and repeated the original error, resulting in subsequent treatments of 150 rads to each side of the head for a total of 300 rads per treatment.

The severity of the erythema increased, and after nine treatments at the reduced level, the physician asked for a review of the dose calculations. The recheck identified the error, and the treatments were stopped. The patient had received a total of 8700 rads. The rate of exposure was also significantly greater than that planned.

Cause or Causes - The misadministration occurred as a result of an error by the dosimetrist in calculating the exposure time necessary to provide the radiation dose prescribed by the physician, coupled with a similar error by the second dosimetrist. The errors resulted in an exposure 45% greater than that prescribed, and in an exposure rate about 80% greater than that prescribed.

The dosimetrists' errors would likely have been detected if the standard hospital practice had been followed and another qualified staff member had rechecked the calcuations used in determining exposure times. However, this procedure was not followed in this instance.

A review by an NRC inspector of dose calculations for radiation therapy for other patients during the time this misadministration occurred identified numerous additional instances where this recheck procedure had not been followed. Hospital employees interviewed attributed this failure to follow the procedure to an excessive workload due to a recent staff vacancy that had not yet been filled.

The rechecking procedure was not formalized in a written instruction, and it was not part of the requirements imposed by the hospital's NRC license.

Actions Taken to Prevent Recurrence

Licensee - The hospital has revised its operating procedures to formalize the requirement that all dose calculations be checked by a second qualified individual. Radiation technologists who administer the treatments are instructed not to perform more than two treatments without the dose calculation being rechecked.

The hospital is actively seeking another dosimetrist to bring the number of dosimetrists to the normal complement of three. The hospital has also instituted an audit program for a periodic review of the radiation therapy department activities by a qualified hospital member from outside the department.

The hospital is providing continuing medical review of the patient.

NRC - The NRC retained a medical consultant to evaluate the misadministration. A special inspection was conducted by the NRC on March 12-13, 1984, to review the circumstances of the misadministration. A meeting between hospital personnel and the NRC staff was held April 3, 1984, to review the hospital's corrective actions as a result of the misadministration. A followup inspection was conducted on April 5-6, 1984, to review the corrective actions being taken.

The licensee prepared a teletherapy treatment Quality Assurance Program Outline and submitted it to the NRC for review and approval on April 17, 1984. The program was written to provide enhanced assurance that all calculations for treatment with the cobalt-60 teletherapy unit are accurately made and verified by an independent dosimetrist and that licensed material is safely used. On July 17, 1984, the NRC issued a Confirmatory Order, effective immediately, for the licensee to implement the program if it has not already been implemented. The NRC will review the effectiveness of the program during subsequent inspections.

This incident is closed for purposes of this report.

AGREEMENT STATE LICENSES

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 first calendar quarter of 1984, one of the Agreement States reported the following abnormal occurrence to the NRC.

AS84-1 Overexposure of a Radiographer and Assistant

Appendix A (Example 1 of "For All Licensees") of this report notes that 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 can be considered an abnormal occurrence.

Date and Place - On February 20, 1984, an industrial radiographer and his assistant employed by Industrial NDT, Inc., North Charleston, South Carolina, received significant doses to the whole body and hands. The incident took place at the Westvaco Corporation paper mill in Charleston, South Carolina.

Nature and Probable Consequences - A radiographer's assistant was utilizing a Tech Ops Model 683 exposure device containing 67 curies of iridium-192. The unsupervised assistant was allowed by the radiographer to make exposures of pipe welds on the roof of a chemical recovery boiler while the radiographer was at a ground floor location performing other radiography work.

While performing one of many exposures that evening, the assistant stumbled and experienced a hard fall within a few feet of the exposed source. The fall was severe enough to cause him to become ill (disorientation and vomiting). He retracted the source into the camera and proceeded to set up his next shot.

Upon exposing the source again, the assistant proceeded to a stairway that served as the 2mr/hr barrier on the floor. While he was monitoring the exposure time, he received a call from the radiographer working on the ground floor. The radiographer informed his assistant that the previous shots which the assistant had taken of the roof welds were of poor quality. At this point, the radiographer agreed to come to the roof to offer guidance. The assistant went to the elevator and proceeded to pick up the radiographer on the ground floor. The assistant claimed that he retracted the source before going to the elevator but he was somewhat uncertain about this matter.

Upon returning to the top floor, the radiographer and the assistant went directly to the source and made adjustments to the exposure set-up. Neither the radiographer nor the assistant checked the exposure area with a survey instrument. They both assumed that the source was in the shielded position.

The amount of time spent near the source is uncertain. The best estimate appears to be 2.5 - 5.0 minutes. During that time, both individuals touched the end of the source tube for a number of seconds. If the source was in the exposed position at the end of the guide tube, the dose received by the hands would be sufficient to cause physical damage.

When all adjustments to the set-up were completed, both individuals moved to a position away from the source to begin the next exposure. The radiographer went for the survey meter that was located about 25 ft. away from the source. The assistant went to the camera crankout and begin to move the source to what he thought was the exposed position. At this time, the radiographer noticed that the survey meter was already reading offscale and he immediately called for his assistant to roll the source back into the camera. The assistant reversed his cranking motion and rolled the source until it stopped. He thought it was in the shielded position. However, the survey meter was still reading off scale. The two individuals fled from the source area to the 2mr/hr barrier. They discussed the matter and decided that the source was either still exposed or the guide cable had been broken. The radiographer approached the guide cable crank, noticed that the indicator on the handle read "exposed" and proceeded to roll the source back into the camera until the survey meter was reading background.

At a later time, both individuals noticed that their pocket dosimeters were offscale. They reset the dosimeters and proceeded to finish their evening work. However, the dosimeter log for that day recorded the radiographer's exposure as 40 mr and the assistant's exposure as 60 mr.

When the individuals completed their work the following morning, they discussed the incident with the day-shift supervisor. He suggested that they notify the radiation safety officer (RSO) if they thought an overexposure had occurred. The individuals convinced themselves that an overexposure had not occurred and they dismissed the matter. The day-shift supervisor took no further action concerning the incident.

On or around February 27, 1984, the radiographer's assistant experienced numbness in the fingers of his left hand. The fingers began to hurt on March 2, 1984, and they became swollen and discolored. Doctors at the Charleston County Medical Center diagnosed the symptoms as possibly being the result of frostbite.

On March 5 or 6, the radiographer began to experience symptoms of numbness and hardening of the skin on two fingers of his right hand. The tips of the fingers eventually turned white. The radiographer did not seek medical attention until March 13.

Both individuals reported the matter to the Director of Operations, Industrial NDT, on March 9. He informed the RSO of the potential overexposure on March 10. The filmbadges were sent to the processing company for emergency processing that day. The results of the analysis were received by the licensee on March 12. On this same date, the licensee notified the South Carolina Department of Health and Environmental Control of the overexposure.

The filmbadge reports indicated that the assistant and the radiographer received whole body doses of about 63 rem and 9 rem, respectively. A reenactment of the incident was conducted to determine time and distance values for estimating actual exposure to the individuals. It is estimated that the assistant and the radiographer received a hand exposure of 5348 rads and 3013 rads, respectively.

Both individuals were referred to the Medical University of South Carolina for treatment and observation. All applicable laboratory tests and studies have been performed to ascertain radiological injury.

<u>Cause or Causes</u> - Apparently, failure to conduct a proper physical survey and failure to follow established procedures to remove the source guide tube and plug the radiography camera after each exposure were the causes of the incident. If the physical survey had been conducted by the individuals, as required by regulation and the licensee's operating procedures, the excessive radiation levels would have been detected and the overexposures may have been avoided. In addition, allowing an inexperienced and unsupervised individual to use radiographic equipment contributed equally to this incident.

Actions Taken to Prevent Recurrence

Licensee - The two radiography personnel have been removed from any future involvement with the use of radioactive materials. All radiographic and supervisory personnel have been retrained in radiation safety and operating procedures.

South Carolina Department of Health and Environmental Control - The regulatory agency investigated the circumstances associated with the event. A show cause order will be issued citing all applicable violations and requiring corrective actions. Civil penalties will be assessed as determined by the corresponding severity level for each violation. An upper management conference will be held with the licensee and additional safety requirements will be negotiated.

This incident is closed for purposes of this report.

REFERENCES

1. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 84-39, "Inadvertent Isolation of Containment Spray Systems," May 25, 1984.*

2. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Through Wall Crack in Vent Header Inside BWR Containment Torus," Federal Register, Vol. 49, No. 92, May 10, 1984, 19912-19914.

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

4. Letter from Richard C. Lewis, Director, Division of Project and Resident Programs, NRC Region II, to R. J. Kelly, Executive Vice President, Georgia Power Company, transmitting a Notice of Violation, Docket Nos. 50-321 and 50-366, March 14, 1984.*

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

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

7. Confirmatory Action Letter from James G. Keppler, Regional Administrator, NRC Region III, to J. W. Reynolds, Executive Vice President, Consumers Power Company, Docket No. 50-155, March 3, 1984.*

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

APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

The following criteria for this report's abnormal occurrence determinations were set forth in an NRC policy statement published in the FEDERAL REGISTER on February 24, 1977 (Vol. 43, 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 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 (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 significant relative to normally expected performance and which is judged to be caused by theft or diversion or by substantial breakdown of the accountability system.

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

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

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

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

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

For Commercial Nuclear Power Plants

1. Exceeding a safety limit of license Technical Specifications (10 CFR §50.36(c)).

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

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

4. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or Technical Specifications that requires immediate remedial action.

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

For Fuel Cycle Licenses

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

2. A major condition not specifically considered in the Safety Analysis Report or Technical Specifications that requires immediate remedial action.

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

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the January through March, 1984 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

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; Vol. 5, No. 3; Vol. 5, No. 4; Vol. 6, No. 1; Vol. 6, No. 2; Vol. 6, No. 3; and Vol. 6, No. 4. It is further updated as follows.

Reactor Building Entries

During the first calendar quarter of 1984, 35 entries were made into containment. There have been a total of 347 entries since the March 28, 1979 accident. Major activities included preparation for and the load test of the reactor building polar crane, taking of core debris samples, partial detensioning of the reactor vessel head studs, and video mapping of the reactor vessel internals.

EPICOR-II/Submerged Demineralizer System (SDS) Processing

The EPICOR-II system processed approximately 79,800 gallons of water during the first quarter of 1984. The SDS processed approximately 85,500 gallons of water during the same time period.

EPICOR-II/Prefilter and SDS Liner Shipments

One SDS liner was shipped from the TMI site to Rockwell Hanford, Hanford, Washington, during this reporting period.

Spent Fuel Pool "A" Refurbishment

Due to funding constraints, refurbishment of the "A" spent fuel pool has had limited progress to date. If additional funding can be allocated, operations may resume at a normal pace during the second quarter of 1984.

Auxiliary and Fuel Handling Building Activities (AFHB)

Decontamination of areas necessary to provide access for surveillance of safety-related equipment containued during this quarter. The pace of decontamination activities in the AFHB decreased due to limited operating funds for 1984 and a major emphasis on reactor head lift and defueling activities.

Makeup and Purification Demineralizer Disposal

Preparations continued for the removal of radioactive resins from the makeup and purification demineralizers in late 1984. On February 25, a sample of the contents of the 'A' demineralizer vessel was obtained. A sampling tool was inserted through the resin fill line, forced through the surface crust on the resin bed, and used to successfully withdraw a 60 milliliter sample. About one third of the sample appeared to be resin beads and the remainder was liquid. The sample was similar in appearance to the samples previously obtained from the 'B' demineralizer. The unshielded sample bottle had a contact radiation level of 6 R/hr. When shielded, the highest contact dose rate was 250 mrem/hr, which was reduced to 16 mrem/hr at a distance of one foot. The shielded sample was shipped on February 28 to Oak Ridge National Laboratory for testing to determine its sluicability and the effectiveness of the proposed process for cesium elution. The processing hardware for the cesium removal system (Phase I) was received onsite on March 29, 1984. The equipment will be tested prior to installation. Software for equipment installation and operation is in the preparation, review and approval process. Preliminary engineering for Phases 2 and 3 of the project, which includes sluicing of the spent resins and packaging of the resin for shipping, has begun. Commencement of the cesium removal process is still scheduled for June 1, 1984.

Reactor Building Polar Crane

During the week of February 13, 1984, a six ton internals indexing fixture, four missile shields weighing approximately 40 tons each (located over the reactor vessel) and one missile shield weighing approximately 30 tons (located over the pressurizer) were successfully moved to a staging area away from the reactor. The missile shields were stacked and held together by structural steel to form a load test assembly weighing approximately 214 tons. During the load test on February 29, 1984, the whole assembly was successfully picked up and moved in various directions with strict limits on where the load could travel. After post-test performance checks were made, the missile shields were moved to their normal storage location on a D-ring.

Reactor Vessel Stud: Partial-Stud Detensioning

The reactor vessel head studs were partially detensioned during early March 1984. This partial detensioning was to determine whether any of the stud nuts holding the head in place were stuck beyond the capabilities of normal unbolting techniques. Calculations indicate that after the partial detensioning (which included the full removal of two studs), the pressure retaining capability of the reactor vessel flange will be greater than or equal to 1000 psig. Currently the reactor coolant system is maintained at a pressure between 0 and 100 psig.

Core Debris Sampling/Video Mapping

Five core debris samples were taken from the reactor vessel during the second week of March 1984. The sample probe was inserted into the debris bed as deep as possible using manual force. In the center of the core, the probe came to a firm stop 30 1/2 inches below the top of the debris bed. The stop depth coincided with an inconel spacer grid, located at approximately the 304 feet, 3 inches elevation in the core. Debris samples were also taken through an open control rod drive mechanism midway between the core center and the periphery. In this location, the probe penetrated to elevation 303 feet, 5 3/4 inches, approximately 37 inches below the top of the debris bed. (The top of the debris bed is not a uniformly level surface.)

A comprehensive video mapping of the reactor vessel internals commenced on March 29, 1984 to better assess the actual condition of the TMI-2 core.

EPA "Sentri" Telemetric Radiation Detection System

A teletype remote readout terminal for the Environmental Protection Agency's (EPA) Reuter-Stokes RA 1011 "Sentri" system of pressurized ion chamber radiation detection telemetry units has been installed at the TMI Program Office on site. The 13 detector units are located at radial distances ranging from 0.5 to 3.5 miles from TMI. The central processor is located at the EPA Field Station in Middletown, Pennsylvania.

The system printout indicates the time of the detector chamber reading, unit location data, an integrated dose in millirem per hour for each detector, the integration time, alarm level, and alarm condition. At present, the EPA printing interval is hourly, and date of the year, as well as the hour, minute and second of the print time, are recorded.

Each sample station also includes a continuous charcoal and particulate air sampler and thermoluminescent dosimeter (TLD). The air sampling cartridges are changed at least weekly and analyzed by gamma spectroscopy at the Middletown Field Station. TLDs are changed quarterly and read at EPA's Office of Radiation Protection Facility in Las Vegas, Nevada.

TMI-2 Advisory Panel Meetings

The Advisory Panel obtains local citizens' views and provides the Commission with valuable counsel on the actions to be proposed and taken by the NRC regarding cleanup of the damaged reactor. On January 12, 1984, the Advisory Panel for the Decontamination of Three Mile Island Unit 2 held a meeting in Harrisburg, Pennsylvania. Mr. Philip Clark, President, GPUN Corporation, and Mr. Edwin Kintner, Executive Vice President, GPUN Corporation, provided the Panel with a discussion of funding for 1984. Mr. Clark stated that the company had firm commitments for funding at a level of $75 million. With the $75 million, the company believes that it can make significant useful progress in the cleanup of Unit 2. With additional funds, Mr. Clark felt that additional progress could be made. The company is presently investigating other possible sources of funding, including customer revenues, and the Edison Electric Institute.

On February 3, 1984, the Panel met internally for a short meeting at 10:00 a.m. and then met with the NRC Commissioners at 11:00 a.m. at the NRC's Washington Office. At the meeting with the NRC Commissioners, the Panel first inquired about the status of the NRC staff's review and response to the NRC Office of Investigations report on the allegations made by Mr. Parks and Mr. Gischel. The NRC Chairman informed the Panel that the Commission had recently received the staff's response and it is presently under review. Dr. B. Snyder provided a short summary of the staff's findings.

The Panel recommended to the Commission that the NRC not consider the restart of Unit 1 until total funds are committed to the cleanup of Unit 2.

The Commission and the Panel discussed the issue of alternate disposal of the processed accident water presently stored on the island. The Panel was informed that the water is presently being used and recycled in the licensee's decontamination efforts, and will continue to be used throughout defueling and decontamination of the damaged reactor. The Panel was informed by the Commission that it is the NRC staff's belief that a decision on the disposal of this water is several years away and that no decision is needed at this time. The Chairman of the Commission assured the Panel that when the time came to make the decision on the method of disposal, the public's concerns would be factored into the NRC's decision.

On February 9, 1984, the Panel held a meeting in Harrisburg, Pennsylvania. GPUNC, the Pennsylvania Department of Environmental Resources, the NRC and the EPA provided presentations on their respective radiation monitoring programs in the TMI area. Mr. Glen Sjoblom, Director of EPA's Office of Radiation Programs, addressed the Panel on EPA's future role in radiation monitoring at TMI. He requested that the other State and Federal agencies involved in monitoring meet with EPA and reexamine the total monitoring program to determine if redirection is warranted and if there is any unnecessary monitoring. The Panel took the positions that, 1) EPA's effort at TMI is essential to the radiation monitoring program in the TMI area and should not be phased out, and 2) EPA should convene a meeting of the State and Federal groups that conduct radiation monitoring in the area to determine if any changes to the program are warranted. The Panel requested that EPA notify the Panel of the results of any such meeting and any recommendations issuing from the meeting relative to changes in the monitoring program.

The topic of discussion for the second half of the Panel meeting was radiation health effects. The NRC and GPUNC provided a panel of experts on the health effects of low level ionizing radiation. The discussion centered around the risk of health effects associated with the estimated occupational radiation exposures due to the Unit 2 cleanup effort.

Future reports will be made as appropriate.

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81-7 Blockage of Coolant Flow to Safety-Related Systems and Components

This abnormal occurrence was originally reported in NUREG-0090, Vol. 4, No. 4, "Report to Congress on Abnormal Occurrences: October-December 1981." It is updated as follows.

Service water systems of nuclear power plants are typically open cycle systems. An open cycle service water system implies that water is pumped directly from a river, cooling pond, or ocean body into the service water intake structure. An immediate problem associated with open cycle systems is that along with water, a variety of mud, silt, sand, algae, bacteria, fungi, and aquatic organisms are also pumped into the service water systems. Although gratings, screens, and filters block out much of the impurities, fouling of service water systems is an existing problem that must be satisfactorily resolved.

Over the past few years, the NRC staff has been following a number of events that have originated from within the service water systems at operating plants. Many of these events have been caused by system fouling, due to mud, silt, or aquatic organisms. Fouling, which has been allowed to go unchecked due to inadequate preventive maintenance and surveillance programs, has led to degradation of safety-related equipment, forced plant shutdowns, power reductions for repairs and modifications, and overall degraded modes of operation. Although service water system fouling is a serious concern from an operations standpoint, the NRC has no knowledge of a service water system event directly inducing a primary system transient. Other events have been caused by corrosion of pipes and components by brackish or salty water.

System fouling or corrosion represent forms of common mode failure that affect both of the redundant safety-related service water trains and do not offer a straightforward solution. Service water systems among various plants are subject to wide ranges in hyraulics, operating temperatures, materials of construction and physical location. In addition, the system components are generally manufactured by a large number of suppliers.

Safety-related service water systems, which already have separate and redundant piping systems, share the same intake structure and ultimate heat sink. Thus, they share the same potential for common mode failure due to service water impurities. As long as plants use the same ultimate heat sink for the redundant service water trains, it is believed that the potential for common mode failure will always be present. Control strategies must be developed to deal with this concern.

In an attempt to identify such control strategies, the NRC staff has developed Generic Issue 51, "Improving Reliability of Open Cycle Service Water Systems." This Generic Issue outlines a multi-year program to study the causes of service water system fouling, their means for prevention and finally developing a set of recommendations.

Further progress on the problem will be reported bi-annually in NUREG-0933, "A Prioritization of Generic Safety Issues" (Ref. B-1).

This incident is closed for purposes of this report.

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81-8 Seismic Design Errors at Diablo Canyon Nuclear Power Plant

The following abnormal occurrence was originally reported in NUREG-0090, Vol. 4, No. 4, "Report to Congress on Abnormal Occurrences: October-December

1981," and updated in subsequent reports in this series, i.e., NUREG-0090, Vol. 5, No. 1 and Vol. 5, No. 3. It is further updated as follows.

In NUREG-0090, Volume 4, No. 4, it was reported that an equipment layout drawing for Diablo Canyon Unit 2 was used in the analysis and design of the containment annulus steel support structure of Unit 1, which is a mirror image of Unit 2. This resulted in the suspension of the lower power license of Unit 1 in November 1981, and initiation of a design verification program. In NUREG-0090, Volume 5, No. 1 and No. 3, a description and status of the design verification program were provided. The program consisted of two complementary efforts: an Independent Design Verification Program (IDVP) performed by outside companies and an Internal Technical Program (ITP) performed by the licensee (Pacific Gas & Electric Company) and the Bechtel Corporation. The program included a complete seismic design review of all safety-related structures, systems, and components, and a non-seismic design review of three major systems, which encompassed all engineering disciplines.

The design verification has been essentially completed. Since August 1983, Supplements 18 (Ref. B-2), 19 (Ref. B-3) and 20 (Ref. B-4) to the Diablo Canyon Safety Evaluation Report have been issued. The supplements present the staff's conclusion that the IDVP has met the requirements and objectives of the design verification effort, that the design verification efforts by the IDVP and the licensee have identified the significant design deficiencies, and that appropriate corrective actions have been taken to ensure that the design of Diablo Canyon Unit 1 conforms to the licensing criteria. At this time, the licensee has completed efforts on a few remaining issues identified in Supplement 20 to the Safety Evaluation Report which were to be resolved prior to issuance of a full power license.

On November 8, 1983, the Commission reinstated the authority to load fuel and on April 13, 1984, the authority for criticality and low power operation.

Since late 1983, the NRC has received numerous allegations regarding the design, construction, management, and operation of the Diablo Canyon Nuclear Power Plant. While many of the allegations are identical or similar, each one has been identified separately, resulting in approximately 500 allegations. The NRC staff is evaluating, inspecting, and investigating each allegation as appropriate. Supplement 21 (Ref. B-5) and updated by Supplement 22 (Ref. B-6) to the Safety Evaluation Report were issued in December 1983 and March 1984, respectively. These supplements presented the NRC staff's evaluation of the first 200 allegations. The staff performed an evaluation of all allegations in sufficient detail to conclude that none is of sufficient safety significance to preclude low power operation. A number of allegations were identified as requiring resolution prior to exceeding 5% power, most notably certain allegations related to the design and installation of piping and piping supports. This requirement has been made a low power license condition.

Future reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

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

1. Fire at Uranium Storage Facility

On December 7, 1983, a fire seriously damaged a warehouse at the uranium storage facility operated by Edlow International Inc. in East St. Louis, Illinois. There was no release of radioactive material resulting from the fire.

Edlow International is licensed by the NRC to store refined uranium ore ("yellowcake"), natural uranium hexafluoride, and uranium hexafluoride which has been slightly enriched in the uranium isotope 235. No processing is permitted at the facility. Edlow International provides a storage service for uranium between the various steps of processing of uranium into fuel for nuclear power plants. Yellowcake is shipped and stored in sealed steel drums similar to 55-gallon drums. Uranium hexafluoride is shipped and stored in heavy steel cylinders.

The December 7, 1983, fire damaged the portion of the warehouse where the yellowcake was stored, but the drums sustained only superficial damage and no uranium was released. The uranium hexafluoride was stored both in a portion of the warehouse not damaged by the fire and in an outside storage area. The cause of the fire has not been determined conclusively by the East St. Louis Fire Department and the Illinois Fire Marshal.

On April 5, 1984, the NRC staff issued a proposed fine of $1,600 against Edlow International for violations of NRC requirements associated with the fire. The violations included: (1) The automatic sprinkler system was not fully activated by the fire in the east portion of the warehouse; water for the sprinkler system in the center section of the warehouse was shut off; and the shut-off valve for the sprinkler system in the west portion of the warehouse was only partially open; (2) The off-site alarm service did not receive a fire alarm and therefore was unable to notify the fire department (the fire was reported by a passerby); and (3) An automobile and a forklift, containing flammable fuel, were stored in the warehouse, contrary to the requirement that no flammable materials be stored there. The licensee paid the fine and described corrective actions. The NRC will examine the corrective actions during subsequent inspections of the licensee.

Subsequent to the fire, there has been considerable interest by the East St. Louis city government and by area residents in the Edlow facility and its potential hazards.

Yellowcake will not burn, and even if the drums were damaged in a fire, there would be little, if any, spreading of the contamination. Uranium hexafluoride represents a significantly greater chemical hazard than a radiation hazard if released in significant quantities. Uranium hexafluoride is a solid material under normal packaging conditions, but hydrolyzes if exposed to air to form uranyl oxyfluoride, a soluble particulate, and hydrofluoric acid, a strong acid.

To minimize the potential of a fire damaging the uranium hexafluoride cylinders and causing a release of the material, the NRC has issued amendments to Edlow International's license requiring that all uranium hexafluoride cylinders be stored outside and that combustible and flammable substances be restricted from the storage area. The company is also to provide a round-the-clock fire-watch at the site. Further, no additional enriched uranium hexafluoride is to be received at the site until Edlow International completes an assessment of possible relocation or improvements to the existing site.

On December 9, 1983, the NRC's Region III office issued a Confirmatory Action Letter to Edlow International approving the improvements which the company had agreed to make in the fire protection, security, and emergency planning procedures for the facility.

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2. Cobalt-60 Contaminated Steel Incident

On January 16, 1984, a truck driver made a wrong turn near the Los Alamos National Laboratory (Los Alamos, New Mexico) and his truck passed radiation monitors near the roadway. Alarms were actuated, but the driver drove on apparently unaware of what the alarms meant. Later the truck was located and its contents were surveyed. It was found that the truck was carrying steel reinforcing rods (called rebar), used in concrete construction, which were contaminated with radioactive material (cobalt-60). New Mexico authorities informed the NRC of the discovery on January 17, 1984. The origin of the rebar was subsequently traced to a foundry in Chihuahua, Mexico.

Mexican safety authorities were notified of the contamination problem and they promptly stopped further shipments into the U.S. The Mexican authorities also located the source of the contamination, made radiation surveys to determine the magnitude of the problem, and took steps to isolate the contamination and the contaminated materials.

Investigations have shown that in 1977, a Texas firm exported a teletherapy unit to a clinic in Juarez, Mexico. At the time of export, the cobalt-60 source (which consisted of about 6000 pellets, each one millimeter in diameter and length) contained 1,003 curies of radioactivity. The teletherapy unit was not used by the clinic; it was stored in a warehouse in Juarez.

In November or early December 1983, the teletherapy unit was disassembled and on December 6, 1983, the metal was sold as scrap to a junkyard in Juarez. In the process, the source was ruptured allowing release of some of the cobalt-60 pellets. The source at this time had decayed to about 400 curies. The radio-active pellets contaminated the truck which carried the scrap, the scrap itself, and the junkyard. Subsequently, without realizing the existence of the radioactive contamination, the junkyard sold scrap containing cobalt-60 to foundries in Juarez and Chihuahua, Mexico. The foundries thereafter, without any knowledge of the radioactivity problem, exported contaminated steel products to the U.S. in the form of rebar and cast pedestals for tables used in restaurants and hotels.

To coordinate NRC efforts on the problem, initially a special working group was established consisting of representatives of the Offices of International Programs, State Programs, Nuclear Material Safety and Safeguards, and Inspection and Enforcement. Subsequently, the Office of Inspection and Enforcement was assigned responsibility to manage NRC activities.

With NRC assistance, U.S. Customs began monitoring shipments of steel at the principal commercial entry into El Paso, Texas, from Juarez on February 14, 1984. Later this monitoring effort was expanded to include all traffic at the three main points of entry into El Paso.

Three distributors of rebar in El Paso, Texas, and one in Phoenix, Arizona, were identified and customer lists were obtained by state authorities for the period in which potentially contaminated steel could have been shipped. In each state, the state radiation safety authority assumed responsibility for locating the rebar and surveying it for radioactive material. Contaminated rebar was found in six states--Texas, New Mexico, Arizona, Colorado, California, and Nevada. All radiation surveys have been completed and the contaminated rebar segregated and held for shipment back to Mexico. Generally, the truckloads of rebar had radiation levels of from 10-15 millirem/hr with a maximum reading on one truck of 600 millirem/hr. The NRC provided radiation protection guidance for allowing contaminated rebar to remain in structures where concrete had already been poured.

In regard to the table pedestals, the NRC determined that about 1400 customers needed to be surveyed. The customers were located in all 50 states, the Bahamas, Canada, and Singapore. The NRC Region III office distributed the names of customers with potentially contaminated castings to the state radiation safety authorities and requested each state to survey the castings and report the results to the appropriate NRC Regional Office. The NRC Office of International Programs advised the foreign countries of the danger. Guidance concerning the radiation surveys was also provided. Tables with contaminated pedestals were located in 40 states. Contact radiation readings on the pedestals ranged from 25 microrems/hr to a few millirems/hr, with a maximum reading of 375 millirems/hr. At the time the radiation surveys were made, most of the tables were still stored in warehouses. Only a relatively few were in use at restaurants or hotels.

Ground and aerial surveys of El Paso and vicinity indicated that no cobalt-60 pellets from the damaged source had been tracked across the border. Therefore, the contamination entering the U.S. is believed to be limited to that in the rebars and table pedestals. There is no evidence that anyone in the U.S. received a significant exposure from the contaminated steel.

While there was no severe impact on public health and safety in the U.S., there were serious problems in Mexico. Many people were either in contact with or near the cobalt-60 pellets. Some people received serious radiation exposures. Mexican authorities indicated that five persons, including three or more employed at the junkyard, received estimated whole body doses in the range of 100 to 450 rems. One individual, involved with the disassembly of the teletherapy unit, developed skin discolorations on one hand. Radiation levels of 50 rems/hr at 1 meter and 10 millirems/hr at 100 meters were measured emanating from the truck that was used to haul the teletherapy unit to the junkyard. The truck was impounded. Sixty-two of the cobalt-60 pellets were found in the streets of Juarez and other locations in Mexico and cleaned up by the Mexican authorities. A reading of 22 rems/hr at 5 cm was measured from one of the pellets. In addition, an NRC Region V staff member, who was sent to assist the Mexican authorities under the aegis of the Pan American Health Organization, obtained a radiation measurement of 600 rems/hr at 1" from the ground within the pile of scrap in the junkyard.

One of the questions raised as a result of this incident is the proper role of the NRC and other agencies in responding to emergencies of this nature involving contamination by radioactive materials. NRC has subsequently prepared an Interim Plan for NRC Response to Materials Contamination Incidents and will prepare a final plan, including the activities and responsibilities of other agencies, to assure a properly coordinated response to future incidents of this kind.

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3. <u>Main Steam Relief Valve Stuck Open</u>

On March 2, 1984, the reactor at Davis Besse Nuclear Power Station shut down automatically about 12:18 p.m. when a main steam line valve on steam generator (S/G) No. 2 closed. The safety valves associated with S/G No. 2 opened, as designed, to reduce pressure in the secondary steam system (which produces non-radioactive steam from the steam generators to power the turbine generator). However, after pressure reduction, one safety valve failed to close. Davis Besse is a pressurized water reactor, operated by Toledo Edison Company (the licensee), and is located in Ottawa County, Ohio.

Steam continued to escape through this valve, and eventually the low pressure setpoint in the steam and feedwater rupture control system was reached, causing automatic isolation of feedwater to S/G No. 2. All water on the secondary side of the steam generator was converted to steam and was released through the open valve. The secondary side of the steam generator boiled dry in about 5 minutes afer isolation. Reactor cooling was maintained through S/G No. 1 (the stuck valve was on the steam line from S/G No. 2). Steam from the secondary side of this S/G was periodically vented to the atmosphere through the power-operated relief valve to provide cooling.

The safety valve was one of nine associated with S/G No. 2. All other safety valves closed properly. It was later discovered that one had not opened. Efforts by licensee personnel to close the stuck valve were unsuccessful. The valve was replaced about 5 a.m. on March 3, 1984, and the licensee began refilling S/G No. 2. At 9:45 a.m. the licensee opened the main steam isolation valves for both steam generator lines and began directing steam to the condenser for cooling. Following further reductions in reactor pressure and temperature, the licensee at 12:50 p.m. established reactor cooling using the residual heat removal system, the normal shutdown cooling system.

The incident had been declared an Unusual Event (the least severe category in the NRC's emergency classification system) at 12:40 p.m. on March 2, 1984. The Unusual Event classification was terminated at 10:50 a.m. on March 3, 1984.

The NRC's Region III Office in Glen Ellyn, Illinois, dispatched a team of inspectors and supervisors to the Davis Besse site following notification of the incident on March 2. The Region III team monitored the utility's actions in dealing with the incident, replacing the valve, and restoring normal shutdown cooling for the reactor.

Subsequent investigation by the licensee determined that the cause of the valve sticking open was the failure of a cotter pin holding a nut in place on the valve stem. The nut rotated when the valve opened and prevented the safety valve from closing.

The cause of the failure of the safety valve to open has not been determined; it has been disabled and will be repaired or replaced at a later date.

The steam line valve closing, which initiated the incident, was found to have been caused by the failure of a relay, which had not been detected because of a wiring error made during circuit modifications in 1979. The valve closed due to routine surveillance testing on a parallel circuit, which coupled with the relay failure, triggered a signal to close.

On March 3, 1984, NRC Region III issued a Confirmatory Action Letter to the licensee (Ref. C-1), which confirmed the licensee's commitments to investigate the equipment failures that occurred and to evaluate the effects of the event on the plant. The licensee responded in a letter dated March 6, 1984 (Ref. C-2). The plant resumed operations on March 8, 1984.

On April 20, 1984, the NRC issued Inspection and Enforcement Information Notice No. 84-33 to alert licensees to a failure mode of safety and safety/relief valves caused by failed cotter pins (Ref. C-3).

The event had no impact on public health or safety and, therefore, is not reportable as an abnormal occurrence. However, the event did generate considerable media and public interest. This was partly due to the steam emitting from the safety valves to the atmosphere coupled with the sirens of the off-site emergency notification system being activated twice about the time the event began. However, neither siren activation was related to the event itself. The first sounding of the siren was for a periodic test and the second sounding was inadvertent.

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4. Contamination Levels Exceeding Environmental Standards

Kerr-McGee Chemical Corporation (the licensee) is holder of Source Material License No. STA 583 issued by the NRC. The license authorizes the possession of an unlimited quantity of thorium at the Rare Earths Facility, West Chicago, Illinois. Production operations under the license ceased in December, 1973.

Over the years, a portion of the wastes from the plant site have been disposed of through discharge into Kress Creek, a tributary of the DuPage River, either via a storm sewer or a drainage ditch. The wastes entered the creek at a point about 0.7 km south of the Kerr-McGee site. The creek flows in a southeasterly direction for about 2 km to its confluence with the West Branch of the DuPage River.

Radioactive contamination along Kress Creek and the DuPage River was detected as a result of an aerial survey in 1977 and later verified by extensive surveys of the West Chicago area undertaken by Argonne National Laboratory in 1977 and 1978 under contract to the NRC. The results of these surveys, which were limited to measurements of surface exposures and dose rates, were reported in NUREG/CR-0413 published in September 1978. Additional surveys of the creek and river were made by the United States Environmental Protection Agency (EPA) (1980) and Oak Ridge Associated Universities (ORAU) under NRC Contract (1981).

Based on these surveys, the NRC staff, in a letter dated December 18, 1981, requested that the licensee submit a plan for the decontamination of Kress Creek and for storage or disposal of the contaminated soil. After discussions with the licensee, further review of existing data on contamination along the creek and consideration of potential changes in EPA and NRC cleanup actions, the staff decided to further assess the radiological contamination in Kress Creek and informed the licensee, in a letter dated June 4, 1982, it was not necessary to take further action in regard to the December 18, 1981, letter and that the staff would further advise Kerr-McGee upon completion of the assessment.

A comprehensive, radiological survey of Kress Creek has now been performed by ORAU under contract to the NRC. The comprehensive radiological survey was specifically designed to determine not only current direct radiation levels, but also the depth distribution of contamination in the creek and river beds and in bank soils along the creek and river. This survey indicated that lands adjacent to Kress Creek and the West Branch of the DuPage River are contaminated with thorium and with daughter products of the thorium decay chain essentially in secular equilibrium. Soil contamination levels and direct levels of radiation were found to be relatively constant throughout the length of Kress Creek, and to extend downstream along the West Branch of the DuPage River.

The average concentrations of total thorium (Th-232 + Th-228) at 1 meter from the edge of the creek at various depths were: 26.1 pCi/g (picocuries per gram) at the surface; 40.2 pCi/g at 15 cm depth; 38.9 pCi/g at 30 cm depth; 28.9 pCi/g at 60 cm depth; and 18.7 pCi/g between 60 and 90 cm. The soil concentrations decreased with increasing distance from the creek. The highest level of total thorium measured in a sample was 555 pCi/g, with a number of other samples exceeding 200 pCi/g. Many of the highest levels were detected in areas near the storm sewer outfall, and hence constitute a potential source of continuing contamination for locations further downstream.

Direct levels of radiation measured at 1, 5, 10, and 25 meters from the edge of the creek and 1 meter above ground surface averaged 28, 25, 21 and 14 uR/hr (microroentgen per hour) respectively. However, radiation levels greater than 100 uR/hr were detected in several locations. Normal background radiation levels in this area averaged 8.6 uR/hr.

The contamination levels found along the creek exceed the environmental standards promulgated by EPA under authority of the Atomic Energy Act of 1954, as amended, for unrestricted use of areas on which thorium processing wastes have been disposed, as described in 40 CFR §192.41 (48 FR 45947). The NRC is charged with implementation and enforcement of these standards, as described in Section 275d of the Atomic Energy Act of 1954, as amended. The contamination levels also exceed the identical standards established for cleanup of vicinity properties under Title I of the Uranium Mill Tailings Radiation Control Act of 1978, as amended, and published in 40 CFR Part 192, Subpart B. The EPA has stated that these standards are appropriate for cleanup of offsite vicinity properties (Ref. C-4). In each case, the EPA standards were established under a statutory directive to establish standards of general application for the protection of public health, safety, and the environment from the radiological hazards associated with processing of thorium processing waste.

Accordingly, the NRC staff concluded that cleanup of the offsite vicinity properties along Kress Creek and the DuPage River is required and that the following levels of contamination specified in EPA standards are to be used as criteria for the offsite properties:

1. 5 picocuries of radium per gram of soil (pCi/g), averaged over the first 15 centimeters (cm) below the surface, and

2. 15 pCi/g averaged over 15 cm thick layers more than 15 cm below the surface.

The specified levels of contamination may be averaged over areas of 100 square meters.

On March 2, 1984, NRC issued an Order (Ref. C-5) requiring the licensee to show cause why it should not be required to take the following actions:

1. Prepare a remedial action plan for the cleanup of radiologically contaminated areas in and along Kress Creek and the West Branch of the DuPage River and for the subsequent safe storage or disposal of contaminated soil.

2. By July 2, 1984, submit the plan to the Office of Nuclear Material Safety and Safeguards, United States Nuclear Regulatory Commission, for review and approval.

3. After approval by the Office of Nuclear Material Safety and Safeguards, execute the cleanup plan in an expeditious manner.

4. In both the planning and execution of remedial actions, priorities shall be established based on the extent of public exposure resulting from the contamination and the timing of projected disposal or safe storage capacity.

In response to the Order, the licensee has filed an Answer and Demand for Hearing denying the allegations contained in the Order. An Order designating the time and place of hearing will be issued by the Commission.

The event is not considered reportable as an abnormal occurrence. It is being reported here under Appendix C due to the general public interest associated with contamination of areas in the public domain.

REFERENCES
(FOR APPENDICES)

B-1 U.S. Nuclear Regulatory Commission, "A Prioritization of Generic Safety Issues," USNRC Report NUREG-0933, published December 1983*, with supplements to be issued.

U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Docket Nos. 50-275 and 50-323, USNRC Report NUREG-0675, Supplements issued as follows:

B-2 Supplement No. 18, issued August 1983.*

B-3 Supplement No. 19, issued October 1983.*

B-4 Supplement No. 20, issued November 1983.*

B-5 Supplement No. 21, issued December 1983.*

B-6 Supplement No. 22, issued March 1984.*

C-1 Confirmatory Action Letter from James G. Keppler, Regional Administrator, NRC Region III, to Richard P. Crouse, Vice President, Nuclear, Toledo Edison Company, Docket No. 50-346, March 3, 1984.**

C-2 Letter from Richard P. Crouse, Vice President, Nuclear, Toledo Edison Company, to James G. Keppler, Regional Administrator, NRC Region III, Docket No. 50-346, March 6, 1984.**

C-3 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 84-33, "Main Steam Safety Valve Failures Caused by Failed Cotter Pins," April 20, 1984.**

C-4 U.S. Environmental Protection Agency. "Final Environmental Impact Standards for the Control of Byproduct Material from Uranium Ore Processing (40 CFR Part 192)," EPA 520/1-83-008-2, September 1983, Page A.1-3, Comment 6. Also, Federal Register notice, published October 7, 1983 (48 FR 45940).

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*Available in NRC Public Document Room, 1717 H Street, NW., Washington, DC 20555, for inspection. Available for purchase from NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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NRC FORM 335
(6-83)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)

NUREG-0090
Vol. 7, No. 1

2. Leave blank

3. TITLE AND SUBTITLE

Report to Congress on Abnormal Occurrences
January - March 1984

4. RECIPIENT'S ACCESSION NUMBER

5. DATE REPORT COMPLETED

MONTH	YEAR
July	1984

6. AUTHOR(S)

7. DATE REPORT ISSUED

MONTH	YEAR
July	1984

8. 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

9. PROJECT/TASK/WORK UNIT NUMBER

10. FIN NUMBER

11. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

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

12a. TYPE OF REPORT

Quarterly

12b. PERIOD COVERED (Inclusive dates)

January - March 1984

13. SUPPLEMENTARY NOTES

14. 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 January 1 to March 31, 1984

During the report period, there were three abnormal occurrences at the nuclear power plants licensed by the NRC to operate. The first involved an inoperable containment spray system; the second involved a through wall crack in a vent header inside a BWR containment torus; and the third involved a serious degradation of a reactor depressurization system. There were two abnormal occurrences for the other NRC licensees. The first involved an overexposure to a member of the public; and the second involved a therapeutic medical misadministration. There was one abnormal occurrence reported by the Agreement States; the event involved an overexposure of a radiographer and assistant.

The report also contains information updating some previously reported abnormal occurrences

15a. KEY WORDS AND DOCUMENT ANALYSIS

15b. DESCRIPTORS

Containment Spray System; Indian Point Unit 2; BWR Containment Torus; Vent Header; Through Wall Crack; Hatch Unit 2; Reactor Depressurization System; Stuck Valves; Air Amplifier System; Valve Thermal Binding; Big Rock Point; Radiation Overexposure; Radiation Therapy Device; Medical Institution; Therapeutic Medical Misadministration; Industrial Radiographer; Exposed Source; Fire; Uranium Storage Facility; Contaminated Steel from Mexico; Main Steam Relief Valve; Davis Besse; Contamination of Waterway Banks.

16. AVAILABILITY STATEMENT

Unlimited

17. SECURITY CLASSIFICATION (This report)

Unclassified

18. NUMBER OF PAGES

19. SECURITY CLASSIFICATION (This page)

Unclassified

20. PRICE

$

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

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NUREG-0090, Vol. 7, No. 1

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES
JANUARY - MARCH 1984

JULY 1984